PROCEEDINGS OF THE THIRD INTERNATIONAL CONFERENCE
ON THE PEACEFUL USES OF ATOMIC ENERGY

Volume 13

NUCLEAR SAFETY

ACTES DE LA TROISIÈME CONFÉRENCE INTERNATIONALE
SUR L'UTILISATION DE L'ÉNERGIE ATOMIQUE À DES FINS PACIFIQUES

Volume 13

SÛRETÉ NUCLÉAIRE

ТРУДЫ ТРЕТЬЕЙ МЕЖДУНАРОДНОЙ КОНФЕРЕНЦИИ
ПО ИСПОЛЬЗОВАНИЮ АТОМНОЙ ЭНЕРГИИ В МИРНЫХ ЦЕЛЯХ

Том 13

ЯДЕРНАЯ БЕЗОПАСНОСТЬ

ACTAS DE LA TERCERA CONFERENCIA INTERNACIONAL
SOBRE LA UTILIZACIÓN DE LA ENERGÍA ATÓMICA CON FINES PACÍFICOS

Volumen 13

SEGURIDAD NUCLEAR

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EXPLANATORY NOTE

The Proceedings of the Third International Conference on the Peaceful Uses of Atomic Energy comprise a single, multilingual publication of sixteen volumes. This form was prescribed by the General Assembly of the United Nations in approving the Conference budget.

Papers accepted for consideration at the Conference are accordingly printed herein only in the original language of submission, each being followed by its abstract in the other three languages of the Conference.*

The budgetary arrangements for the Conference required also that Governments provide abstracts and papers in two of the Conference languages. One of the three abstracts following each paper is, therefore, in a translation provided by the Government concerned. The abstracts were translated into the other two languages either by the Division of Language Services, International Atomic Energy Agency (IAEA) in Vienna, or, with its assistance, through the intermediary of the national atomic energy authorities in London, Paris, Moscow and Madrid.

The Foreword by the Secretary-General of the United Nations, the Preface by the Director General of IAEA, and this Explanatory Note, together with the records of discussion at each of the six scientific general sessions and thirty-six technical sessions of the Conference, are published in all four languages. All other material, which is largely of a formal nature and is confined to Volumes 1 and 16, is published in the language of submission or delivery, followed in the case of French, Russian and Spanish originals by the English translation.

Governments whose national tongue is not one of the four Conference languages were consulted as to their preference for the language in which their papers should appear in these Proceedings.

The Table of Contents in each volume gives the titles of papers in the original language, or language of choice, followed in the case of French, Russian and Spanish titles by the English translation.

Starting from the 992 abstracts submitted by Governments, specialized agencies and IAEA, the Scientific Secretariat, working under the guidance of the United Nations Scientific Advisory Committee, finally chose 747 papers for inclusion in the Programme of the Conference; of these, 358 were selected for oral presentation at the 42 working sessions.

In arranging the programme, the Scientific Secretariat aimed at achieving a balanced schedule, providing for the oral presentation of as many papers as possible at each session while still leaving adequate time for discussion of the material presented. Two afternoons were left entirely free, to enable informal groups to discuss matters arising out of discussions at the formal sessions of the Conference. No records were taken of such informal meetings.

Wherever possible, the author, or authors, of papers were consulted during the Conference by members of the Scientific Secretariat, who acted as secretaries of session, or by the team of editors made available for the purpose by IAEA,** to ensure maximum accuracy.

The records of discussion at the various sessions, based on notes taken in the meetings by IAEA records officers,** and checked where necessary against the sound recordings made of all sessions, were prepared by the Division of Language Services of IAEA in English, and subsequently translated into French, Russian and Spanish through the intermediary of the atomic energy authorities in the three countries concerned (see third paragraph of the present note).

The editing of the English, French and Spanish papers was carried out at the United Nations Office at Geneva under United Nations supervision by a team of editors, whose services, also, were made available by the atomic energy authorities of their respective countries, with some help from outside consultants. The editing of the Russian papers was done in Moscow in similar circumstances. The following served as editors: Mr. A. de Calmès, Dr. C. E. Granados, Mr. D. H. Hill, Mr. V. F. Kalinin, Cand. Tech. Sc., Dr. R. Lapage, Mr. E. T. Marles, Dr. J. D. C. Mole, Mr. C. Ségot, Mr. J. J. Stobbs, Mr. C. R. Symons and Mr. J. Williamson.

The task of printing this large collection of scientific information has been shared by printers in

* The languages of the Conference were English, French, Russian and Spanish.

** The names of the scientific secretaries, editors and records officers will be found in the list of the Conference Secretariat in Annex 1, Volume 1, of this series.

Belgium, Canada, France, Switzerland, the Union of Soviet Socialist Republics and the United Kingdom.

Full titles of the sixteen volumes of these Proceedings, together with the sessions covered by each volume, are as follows:

Volume No.		Sessions included
1	Progress in Atomic Energy	A, B, 1.6, C, H
2	Reactor Physics	3.1
3	Reactor Studies and Performance	3.2, 3.3
4	Reactor Control	3.4, 3.5
5	Nuclear Reactors — I. Gas-cooled and Water-cooled Reactors	1.1, 1.2, 1.3
6	Nuclear Reactors — II. Fast Reactors and Advanced Concepts	1.4, 1.5, 1.7
7	Research and Testing Reactors	D, 1.9, 1.8
8	Reactor Engineering and Equipment	1.10, 1.11, 3.7
9	Reactor Materials	2.8, 2.9, 2.4
10	Nuclear Fuels — I. Fabrication and Reprocessing	2.3, 2.6, 2.7
11	Nuclear Fuels — II. Types and Economics	2.5, 2.1, 2.2
12	Nuclear Fuels — III. Raw Materials	2.11, 2.12, 2.10
13	Nuclear Safety	3.9, 3.8 3.6
14	Environmental Aspects of Atomic Energy and Waste Management	3.10, 3.11
15	Special Aspects of Nuclear Energy and Isotope Applications	E, 4.1, F, G, 4.2
16	List of Papers and Indexes	

NOTE EXPLICATIVE

Les Actes de la troisième Conférence internationale sur l'utilisation de l'énergie atomique à des fins pacifiques sont publiés ici sous la forme d'une édition unique, multilingue, en seize volumes. Cette présentation a été décidée par l'Assemblée générale lorsqu'elle a approuvé le budget de la Conférence.

En conséquence, les mémoires qui ont été acceptés pour la Conférence sont reproduits ici dans la langue originale dans laquelle ils ont été soumis et sont suivis d'un résumé dans les trois autres langues de la Conférence *.

Aux termes des dispositions budgétaires prises en vue de la Conférence, les gouvernements devaient fournir les résumés et les mémoires dans deux des langues de la Conférence. Ainsi, sur les trois résumés qui suivent chaque mémoire, un est une traduction fournie par le gouvernement intéressé. La traduction des résumés dans les deux autres langues a été faite soit par la Division des services linguistiques de l'Agence internationale de l'énergie atomique (AIEA), à Vienne, soit avec son concours, par les soins des organismes nationaux compétents en matière d'énergie atomique à Londres, Paris, Moscou et Madrid.

L'avant-propos du Secrétaire général de l'Organisation des Nations Unies, la préface du Directeur général de l'AIEA et la présente note explicative, ainsi que les comptes rendus de chacune des six séances scientifiques générales et des trente-six séances techniques de la Conférence, sont publiés dans les quatre langues. Tous les autres textes, qui pour la plupart sont d'un caractère non technique et figurent dans les volumes 1 et 16, sont publiés dans la langue dans laquelle ils ont été présentés par écrit ou oralement et sont suivis, lorsque cette langue est l'espagnol, le français ou le russe, d'une traduction en anglais.

Les gouvernements des pays dont la langue officielle n'est pas l'une des quatre langues utilisées à la Conférence ont été consultés pour savoir dans quelle langue ils préféraient voir paraître leurs mémoires.

La table des matières de chaque volume donne les titres des mémoires dans la langue originale ou dans la langue choisie; ces indications sont suivies, pour les titres en espagnol, en français et en russe, de la traduction en anglais.

Sur les 992 résumés présentés par les gouvernements, les institutions spécialisées et l'AIEA, le Secrétariat scientifique, travaillant sous la direction du Comité consultatif scientifique des Nations Unies, en a finalement retenu 747 pour les inscrire au programme de la Conférence; sur ce nombre, 358 ont été présentés oralement aux 42 séances de travail.

En établissant le programme de la Conférence, le Secrétariat scientifique a cherché à réaliser un équilibre: il s'est efforcé de ménager un temps suffisant pour la présentation du plus grand nombre possible de mémoires tout en laissant du temps pour leur discussion. Deux après-midi avaient été laissés entièrement libres afin de permettre aux participants d'organiser des réunions non officielles et de discuter en petits groupes des questions qui se posaient à la suite des séances officielles de la Conférence. Ces réunions n'ont pas fait l'objet de comptes rendus.

Toutes les fois que cela a été possible, l'auteur ou les auteurs des mémoires ont été consultés pendant la Conférence par les membres du Secrétariat scientifique, qui ont assuré le secrétariat des séances, ou par l'équipe d'« éditeurs » que l'AIEA ** avait mis à cet effet à la disposition de la Conférence, afin d'assurer l'exactitude la plus grande.

Les comptes rendus des discussions aux réunions, établis d'après les notes prises en séance par les rédacteurs de comptes rendus de l'AIEA** et comparés toutes les fois qu'il le fallait avec les enregistrements sonores, ont été rédigés en anglais par la Division des services linguistiques de l'AIEA, puis traduits en espagnol, en français et en russe par les soins des organismes compétents en matière d'énergie atomique des trois pays intéressés (voir le troisième alinéa de la présente note).

Les mémoires rédigés en anglais, en espagnol et en français ont été mis au point pour l'impression à l'Office européen des Nations Unies à Genève, sous le contrôle de l'ONU, par une équipe de rédac-

* Les langues de la Conférence étaient l'anglais, l'espagnol, le français et le russe.

** On trouvera les noms des secrétaires scientifiques, des « éditeurs » et des rédacteurs de comptes rendus dans la liste des membres du secrétariat de la Conférence à l'annexe 1 du volume 1.

teurs mis à la disposition de la Conférence par les organismes compétents en matière d'énergie atomiques des pays intéressés, avec l'aide de quelques consultants extérieurs. La mise au point définitive des mémoires rédigés en russe a été faite à Moscou dans les mêmes conditions. Voici les noms des rédacteurs qui ont assuré la mise au point des mémoires: M. A. de Calmès, M. C. E. Granados, M. D. H. Hill, M. V. F. Kalinin, M[lle] R. Lapage, M. E. T. Marles, M[lle] J. D. C. Mole, M. C. Ségot, M. J. J. Stobbs, M. C. R. Symons et M. J. Williamson.

Des entreprises de Belgique, du Canada, de France, du Royaume-Uni, de Suisse et de l'Union des Républiques socialistes soviétiques se sont partagé la tâche que représentait l'impression de cette masse importante de documents scientifiques.

Les titres complets des seize volumes des Actes de la Conférence, ainsi que les numéros des séances sur lesquelles porte chaque volume, figurent ci-après:

Numéro du volume		*Séances*
1	Progrès accomplis dans le domaine atomique	A, B, 1.6, C, H
2	Physique des réacteurs	3.1
3	Etude des réseaux et performance des réacteurs	3.2, 3.3
4	Contrôle des réacteurs	3.4, 3.5
5	Réacteurs nucléaires — I. Réacteurs refroidis par un gaz et réacteurs refroidis à l'eau	1.1, 1.2, 1.3
6	Réacteurs nucléaires — II. Réacteurs à neutrons rapides et réacteurs d'avant-garde	1.4, 1.5, 1.7
7	Réacteurs de recherche et réacteurs d'essai de matériaux	D, 1.9, 1.8
8	Technologie et équipement des réacteurs	1.10, 1.11, 3.7
9	Matériaux pour réacteurs	2.8, 2.9, 2.4
10	Combustibles nucléaires — I. Fabrication et retraitement	2.3, 2.6, 2.7
11	Combustibles nucléaires — II. Caractéristiques et aspects économiques	2.5, 2.1, 2.2
12	Combustibles nucléaires — III. Matières premières	2.11, 2.12, 2.10
13	Sûreté nucléaire	3.9, 3.8, 3.6
14	Influence sur le milieu de l'emploi de l'énergie nucléaire. Traitement et élimination des déchets	3.10, 3.11
15	Aspects particuliers de l'énergie nucléaire et applications des radioéléments	E, 4.1, F, G, 4.2
16	Liste des mémoires et index	

ПОЯСНИТЕЛЬНАЯ ЗАПИСКА

Труды третьей Международной конференции по использованию атомной энергии в мирных целях представляют собой единое многоязычное издание из шестнадцати томов. Такая форма была предусмотрена Генеральной Ассамблеей Организации Объединенных Наций при одобрении ею бюджета Конференции.

Принятые к рассмотрению Конференцией доклады соответственно опубликованы здесь лишь на языке оригинала; при этом каждый доклад сопровождается аннотацией на других трех языках Конференции*.

Бюджетные постановления в отношении проведения Конференции также предусматривали, что правительства представят аннотации и доклады на двух языках Конференции. Поэтому одна из трех аннотаций, сопровождающих каждый доклад, является переводом, представленным соответствующим правительством. Аннотации были переведены на другие два языка либо Отделом переводов Международного агентства по атомной энергии (МАГАТЭ) в Вене, либо с его помощью при сотрудничестве национальных органов, ведающих вопросами атомной энергии, в Лондоне, Париже, Москве и Мадриде.

Введение и предисловие Генерального Секретаря Организации Объединенных Наций и Генерального директора МАГАТЭ, соответственно, и настоящая пояснительная записка, наряду с протоколами каждого из шести научных пленарных заседаний и тридцати шести секционных заседаний Конференции, публикуются на всех четырех языках. Все другие материалы, которые по своему характеру в основном относятся к числу официальных и содержатся в томах 1 и 16, публикуются на языке оригинала; и когда речь идет о французских, русских и испанских оригиналах, то к ним приложен английский перевод.

С правительствами стран, язык которых не относится к числу четырех языков Конференции, были проведены консультации по поводу того, на каком языке было бы желательно, по их мнению, опубликовать в настоящих трудах представленные ими доклады.

В содержании каждого тома указаны заглавия докладов на языке оригинала либо на другом избранном языке, и в том случае, когда речь идет о французских, русских и испанских заглавиях, их сопровождает английский перевод.

Из 992 аннотаций, представленных правительствами, специализированными учреждениями, а также МАГАТЭ, Ученый секретариат, работая под руководством Научного консультативного комитета Организации Объединенных Наций, в итоге отобрал 747 докладов для включения их в программу Конференции; из них 358 были отобраны для представления в устной форме на 42 рабочих заседаниях.

При составлении программы Ученый секретариат ставил целью добиться сбалансированного расписания, которое дало бы возможность представить в устной форме максимальное количество докладов на каждом заседании при обеспечении достаточного времени для проведения дискуссии по поводу представленного материала. В двух случаях имеющееся во второй половине дня время оставили нераспределенным, с тем чтобы дать возможность неофициальным группам обсудить вопросы, возникшие в ходе дискуссии на официальных заседаниях Конференции. На таких неофициальных заседаниях протоколы не составлялись.

По мере возможности, с автором или авторами докладов консультировались в ходе Конференции члены Ученого секретариата, которые выполняли функции секретарей заседаний, либо такие консультации проводились группой редакторов, которые были выделены МАГАТЭ** для этой цели, с тем чтобы обеспечить максимальную точность.

* Языками Конференции являлись: английский, французский, русский и испанский.

** Фамилии ученых секретарей, редакторов и протоколистов приведены в перечне сотрудников секретариата Конференции в приложении 1-ом к тому 1-му настоящей серии.

Протоколы дискуссии на различных заседаниях, составленные на основе записей, сделанных в ходе заседаний протоколистами МАГАТЭ*, и проверенные, по мере необходимости, путем сравнения со звуковой записью, которая велась на всех заседаниях, были подготовлены Отделом переводов МАГАТЭ на английском языке и впоследствии переведены на французский, русский и испанский языки при сотрудничестве национальных органов, ведающих вопросами атомной энергии, в трех заинтересованных странах (смотри третий абзац пояснительной записки).

Работа по редактированию документов на английском, французском и испанском языках была проведена в Европейском отделении Организации Объединенных Наций, в Женеве, под руководством Организации Объединенных Наций группой редакторов, услуги которых были также предоставлены по линии органов, ведающих вопросами атомной энергии в соответствующих странах, с использованием в некоторой степени помощи приглашенных со стороны консультантов. Русские документы редактировались в Москве в таких же условиях. Нижеследующие лица осуществляли работу в качестве редакторов: д-р К. Э. Гранадос, кандидат технических наук В. Ф. Калинин, г-н А. де Кальмэс, д-р Р. Лепейдж, г-н Э. Т. Марлз, д-р Дж. Д. К. Моул, г-н Ч. Р. Саймонс, г-н Дж. Дж. Стобз, г-н Ш. Сэго, г-н Дж. Уильямсон, г-н Д. Х. Хилл.

В выполнении задачи по печатанию этой обширной научной информации принимали участие типографии в Бельгии, Канаде, Соединенном Королевстве, Союзе Советских Социалистических Республик, Франции и Швейцарии.

Ниже приводятся полные заглавия шестнадцати томов настоящих Трудов, а также указывается, какие сессии охватываются каждым томом:

* Фамилии ученых секретарей, редакторов и протоколистов приведены в перечне сотрудников секретариата Конференции в приложении 1-ом к тому 1-му настоящей серии.

Номер Тома		Заседания, включенные в том
1	Прогресс в работах по атомной энергии	A, B, 1.6, C, H
2	Физика реакторов	3.1
3	Изучение реакторов и их характеристики	3.2, 3.3
4	Регулирование реакторов	3.4, 3.5
5	Ядерные реакторы — I. Реакторы с водяным и газовым охлаждением	1.1, 1.2, 1.3
6	Ядерные реакторы — II. Реакторы на быстрых нейтронах и усовершенствованные реакторы	1.4, 1.5, 1.7
7	Исследовательские и испытательные реакторы	D, 1.9, 1.8
8	Технология и оборудование реакторов	1.10, 1.11. 3.7
9	Реакторные материалы	2.8, 2.9, 2.4
10	Ядерное топливо — I. Изготовление и переработка	2.3, 2.6, 2.7
11	Ядерное топливо — II. Типы и экономика	2.5, 2.1, 2.2
12	Ядерное топливо — III. Сырьевые материалы	2.11, 2.12, 2.10
13	Ядерная безопасность	3.9, 3.8, 3.6
14	Исследование окружающей среды и удаление радиоактивных отходов	3.10, 3.11
15	Специальные аспекты применения ядерной энергии и изотопов	E, 4.1, F, G, 4.2
16	Список докладов и указатели	

NOTA EXPLICATIVA

Las Actas de la tercera Conferencia Internacional sobre la Utilización de la Energía Atómica con Fines Pacíficos están constituidas por una publicación única y plurilingüe compuesta de dieciséis volúmenes, en conformidad con lo dispuesto por la Asamblea General de las Naciones Unidas al aprobar el presupuesto de la Conferencia.

Por consiguiente, las memorias aceptadas para ser examinadas en la Conferencia sólo figuran impresas en el idioma original en que se presentaron, y cada una de ellas va seguida de un resumen de la misma en los otros tres idiomas de la Conferencia *.

En los arreglos presupuestarios para la Conferencia se dispuso también que los gobiernos tenían asimismo que presentar resúmenes y memorias en dos de los idiomas de la Conferencia. En consecuencia, uno de los tres resúmenes que siguen a cada memoria es una traducción facilitada por el gobierno interesado. Los resúmenes fueron traducidos a los otros dos idiomas, ya por la División de Idiomas del Organismo Internacional de Energía Atómica (OIEA) de Viena, o, con su asistencia, por conducto de las autoridades nacionales de energía atómica de Londres, París, Moscú y Madrid.

La introducción del Secretario General de las Naciones Unidas, el prefacio del Director General del OIEA y la presente nota explicativa, junto con las actas de los debates celebrados en cada una de las seis sesiones científicas generales y las treinta y seis sesiones técnicas de la Conferencia, se publican en los cuatro idiomas. El resto del material, que reviste en su mayoría un carácter oficial y está contenido exclusivamente en los volúmenes 1 y 16, se publica en el idioma en que fue presentado o entregado, seguido para los originales en español, francés y ruso, de la traducción en inglés.

Se consultó a los gobiernos cuyo idioma nacional no es uno de los cuatro idiomas de la Conferencia para saber en cuál de ellos preferían que se publicaran sus memorias en estas Actas.

El índice de cada volumen contiene los títulos de las memorias en el idioma original, o en el idioma elegido, seguidos, cuando se trata de títulos en español, francés y ruso, de la traducción en inglés.

De los 992 resúmenes presentados por gobiernos, organismos especializados y el OIEA, la Secretaría Científica, bajo la dirección del Comité Científico Consultivo de las Naciones Unidas, escogió por último 747 memorias que debían ser incluidas en el programa de la Conferencia; de éstas, 358 fueron seleccionadas para ser presentadas oralmente en las 42 sesiones de trabajo.

Al preparar el programa de actividades, la Secretaría Científica trató de conseguir un justo equilibrio, y así se previó la presentación oral del mayor número posible de memorias en cada sesión, pero dejando todavía tiempo suficiente para examinar la información presentada. Se dejaron dos tardes totalmente libres, a fin de que los grupos oficiosos pudieran examinar las cuestiones que surgieran en las sesiones oficiales de la Conferencia. No se levantó acta de tales reuniones.

Siempre que fue posible, el autor, o los autores, de las memorias fueron consultados en el curso de la Conferencia por miembros de la Secretaría Científica, que actuaron de secretarios de sesión, o por un grupo de editores facilitado a dicho efecto por el OIEA**, a fin de asegurar la máxima exactitud.

Las actas de los debates celebrados en las diversas sesiones, basadas en notas tomadas en las reuniones por redactores de actas del OIEA**, y verificadas siempre que fue necesario mediante las grabaciones efectuadas en todas las sesiones, fueron preparadas por la División de Idiomas del Organismo Internacional de Energía Atómica (OIEA) en inglés, y traducidas después al español, el francés y el ruso por conducto de las autoridades de energía atómica de los tres países interesados (véase el tercer párrafo de la presente nota).

La preparación para la publicación del texto de los documentos en español, francés e inglés se efectuó en la Oficina de Ginebra de las Naciones Unidas, bajo la fiscalización de las Naciones Unidas, por un equipo de editores cuyos servicios fueron también proporcionados por las autoridades de energía atómica de sus respectivos países, con alguna ayuda de consultores del exterior. La preparación para la publicación de los documentos en ruso se efectuó en Moscú en circunstancias análogas. Actuaron de editores las personas siguientes: Sr. A. de Calmès, Dr. C. E. Granados, Sr. D. H. Hill,

* Los idiomas de la Conferencia fueron el español, el francés, el inglés y el ruso.

** Los nombres de los secretarios científicos, editores y redactores de actas figuran en la lista de la Secretaría de la Conferencia, en el anexo 1, volumen 1, de esta serie.

Sr. V. F. Kalinin, Dra. R. Lapage, Sr. E. T. Marles, Dra. J. D. C. Mole, Sr. C. Ségot, Sr. J. J. Stobbs, Sr. C. R. Symons y Sr. J. Williamson.

En la impresión de esta gran recopilación de información científica han participado impresores de Bélgica, el Canadá, Francia, el Reino Unido, Suiza y la Unión de Repúblicas Socialistas Soviéticas.

Los títulos completos de los dieciséis volúmenes de estas Actas, junto con las sesiones comprendidas en cada volumen, son los siguientes:

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The nuclear safety research and development program in the United States

By J. A. Lieberman, H. L. Hamester and P. Cybulskis *

The dominant importance of safety considerations in the location, development, design, construction and operation of nuclear reactors has been a hallmark of the nuclear industry since its inception. The safety record of the industry is irrefutable testimony of the value of this attitude, of the efforts made, and of accomplishments by workers associated with this field. Nevertheless, as the production of electrical energy from nuclear reactors becomes more competitive with the burning of fossil fuels, and the incentive for location of power reactors nearer load centers develops, the need for adequate information related to the safety of reactors will correspondingly increase.

The past three years have seen a major emphasis and acceleration of a broad scale, co-ordinated nuclear safety research and development program in the United States Atomic Energy Commission. Particular emphasis has been placed on engineering scale, integral tests to obtain information related to the possible consequences of postulated severe accidents, the so-called "maximum credible accident", and the performance and reliability of safeguards engineered into reactor systems; first, to prevent such accidents from occurring, and second, to attenuate or contain the consequences of severe accidents. While this emphasis is mentioned, the nuclear safety research and development program in the United States strives to achieve and maintain a proper balance between the knowing of what is involved in reactor accidents, and the understanding of why and how these things happen. Stated another way, the general objectives of the program are to provide a sound technical basis for the location, development, design, construction, and operation of nuclear reactor systems in land, marine, and aerospace environmental applications without undue radiation hazards to the public or to persons engaged in these activities; and to provide increased quantitative assurance to designers, manufacturers, administrators, and the public that this primary objective is being attained. Without dwelling on this latter point, it is simply noted that the related problems of power reactor siting, and public understanding and reassurance regarding the safety of these reactors, supply a significant impetus to our program objectives, particularly the short range program requirements. Similarly, there should also be noted the important and useful working relationship between the program and the AEC Advisory Committee on Reactor Safeguards through its Subcommittee on Safety Research; the US nuclear industry through the Reactor Safety Committee of the Atomic Industrial Forum; and the regulatory arm of the AEC.

* US Atomic Energy Commission, Washington, D.C.

PROGRAM SCOPE

As an index of the extent and depth of the program the current fiscal year (1964) budget is cited below in millions of dollars.

Reactor kinetics	4.2
Chemical reactions	1.5
Reactor containment	1.7
Fast reactor safety	1.1
Environmental studies	1.9
Low and intermediate activity wastes	1.1
High activity wastes	3.0
Gaseous and particulate effluents	0.6
Engineering field tests aerospace systems	11.1
Terrestrial systems	1.5
Nuclear safety analysis and evaluation	0.4
	32.0

Parts of the program are described in some detail in other papers at this conference.

REACTOR KINETICS

The reactor kinetics program consists of both integral core or reactor experiments and laboratory scale investigations of abnormal reactor behavior. The integral core experiments are conducted at the AEC's National Reactor Testing Station in Idaho under the Special Power Excursion Reactor Test (SPERT) program. The Phillips Petroleum Company is responsible for the technical management of this program.

The objective of the SPERT program is to obtain experimental data on nuclear excursion behavior,

reactor instability, and other effects representing abnormal reactor behavior. Analytical studies of the data are conducted to develop an understanding of the various mechanisms that influence reactor dynamic behavior, particularly those mechanisms which are associated with large reactivity changes during a reactor transient.

Four different test reactors are used in the experimental program: (*a*) SPERT I, a tank type reactor, open to the atmosphere with no provision for coolant circulation, (*b*) SPERT II, a pressurized reactor (to 300 psi) designed for heavy water (effect of neutron life-time) experiments at moderate initial powers, (*c*) SPERT III, a fully pressurized reactor (to 2 500 psi) having forced circulation, and capable of power operation up to 60 MW (th) for short periods of time, and (*d*) SPERT IV, a large tank type reactor, similar to SPERT I, but with provision for coolant flow and the physical size to accommodate very large cores.

These reactors favor water technology but the data that are developed can be applied to other systems as well. SPERT I started operations in 1955, SPERT III in 1958, SPERT II in 1960, and SPERT IV in 1962. Concise but complete descriptions of these facilities are provided in published reports.

A large number of transient and static tests (more than 2 500) have been conducted with these facilities to study excursion behavior and instability phenomena up to the threshold of core damage. From such experiments, we learned that most reactors have inherent self-shutdown features so that even severe nuclear excursions can be self-limiting in the power levels they attain and self-terminating in that the power level is reduced safely without the use of control rods. To a certain extent, we are able to predict nuclear excursion behavior for reactors that are somewhat different from those we have tested. This information represents a major contribution to our knowledge of reactor behavior. It has alleviated some of the concern and eliminated much misunderstanding that once existed among reactor experts on the interpretation of the magnitude and consequences of so-called "uncontrolled nuclear excursions".

To provide further insight and understanding of the consequences of and phenomena associated with violent, uncontrolled nuclear excursions, three destructive tests have been carried out in the SPERT I machine.

The last of these was carried out in April 1964 with a slightly enriched UO_2-stainless steel fuel typical of US power reactors. In this test a reactivity insertion of about 3.3 dollars resulted in an excursion period of 1.5 m s. The peak power achieved was about 33 000 MW and the total energy release was approximately 175 MW/s. After the test all the control rods and most of the instrumentation were still operable. A reactivity loss about 7 cents following the test indicated a rupture of only 4 or 5 fuel pins. This test demonstrated once again, even under quite rigorous conditions, the very strong inherent safety factors associated with this type of fuel.

The rest of the SPERT program has now been largely re-oriented to experiments involving oxide cores typical of those used in water cooled nuclear power plants. Transient testing of an oxide core has been initiated in SPERT III starting from high initial power conditions. The kinetics of large oxide fueled cores is to be investigated in SPERT IV and alternative uses for the SPERT II machine are under consideration. Among the possibilities are modification for transient testing of gas-cooled reactors and its use as a fuel melt-down test facility.

Another safety project, directed to the investigation of reactor excursion behavior, was conducted with the KEWB reactor. This program was initiated in 1955 with Atomics International as the operating contractor. The general objective of the KEWB program (Kinetic Experiments on Water Boilers) was the determination of the safety characteristics of small aqueous homogeneous reactors known by the title of "Water Boilers". These are typically small research and training reactors of less than 100 kW steady state operating power levels, which are fueled with 10 to 30 liters of an aqueous uranium solution. They normally operate at temperatures below the boiling point of water and absolute pressures of less than one atmosphere.

The KEWB experimental program was completed in 1960 and a large number of technical reports were published summarizing test data and the concurrently developed analytical techniques. All of the major technical goals originally established for the KEWB program were met. The data generated showed that aqueous homogeneous reactors are extremely safe. The only safety limitation is the inertial pressure of the solution due to acceleration by the production of radiolytic gases. Pressure transients were studied with a series of experiments beyond the maximum expected accident for such reactors and analytic expressions were developed to predict peak pressure as a function of various reactor operating parameters.

FUEL-COOLANT REACTIONS

Argonne National Laboratory has been engaged in the study of metal-water reactions for a number of years to determine the percent and rates of reaction for many different metals and test parameters. A substantial amount of data now exists on the interaction of molten fuel materials with water. Data has been obtained on the chemical reaction behavior of uranium, aluminum, zirconium, stainless steel. Metallic, cermet, and oxide fuels clad with such materials have also been tested. Some typical variables that have been evaluated include the effect of initial metal temperature,

the effect of initial water temperature, and the ratio of surface to volume of sample including the relatively severe case of small particles. One accomplishment of this study has been the development of isothermal rate laws which can be applied to the analysis of melt-down reactions.

Unfortunately, analysis of the metal-water reaction in actual core destruction tests by use of rate laws and percentage of reaction so far derived from laboratory experiments requires that post-mortem measurements of the particle size distributions be made and the degree of reaction inferred therefrom. Otherwise, the correlations are not very satisfactory. There appears to be no practical theoretical approach to the problem of predicting what particle or droplet size will be produced during melt-down. The present tendency is, therefore, to rely heavily on the experimental results obtained from melt-downs in the TREAT reactor. For example, melt-down experiments for study of metal-water interactions with SPERT and SL-1 fuel have been conducted under nuclear excursion conditions by using TREAT. The parametric investigations in TREAT provide considerable data for at least post-accident analysis of a wide range of melt-down effects. However, TREAT is limited to reactor periods of 40 ms or longer, and the experimental test volume is too small to allow mock-up experiments of large fuel clusters. Extension of accident effects studies to larger fuel sub-assemblies subjected to short damaging nuclear excursions will be possible with the development of an advanced pulsed neutron reactor which has been named the Power Burst Facility. The conceptual design and requirements for this reactor are being reviewed at the present time.

The Power Burst Facility will permit experimental study of the microscopic and gross effects associated with slow or rapid heating up to melt-down of reactor fuels subjected to a power transient associated with a reactor period as short as one m s. Scaling effects can be studied directly by melting small samples initially and extending the tests to include large fuel clusters having mass-to-coolant ratios and geometric configurations equivalent to those of a given full scale reactor. Measurement of transient void, pressure, and temperature in the test assembly during a power burst will permit calculation of the reactivity worth of the sub-assembly during the transient. For experiments conducted up to fuel melt-down, the mode and rate of energy transfer from the fuel to the coolant will be studied and in the case of pre-irradiated fuel test specimens, fission product release measurements will be made.

FISSION-PRODUCT BEHAVIOR

Increasing attention has been given to studies which investigate the accidental release of fission products due to reactor fuel destruction. The programs in this category have been expanded fourfold over those in existence between 1950 to 1960. Most of the research work on fission-product behavior is conducted at the Oak Ridge National Laboratory. Smaller programs in this area are conducted at Brookhaven National Laboratory and Atomics International.

Previous work has identified the major parameters which control the extent of release, while the current work is emphasizing studies on the behavior of accident-released fission products. Inasmuch as the transport and plateout characteristics of the fission products are determined largely by their physical and chemical nature, a major effort is now under way to characterize the physical and chemical forms of the fission products which would result from a reactor accident.

Parameters such as fuel type, cladding, atmosphere, pressure, peak temperature, time at temperature, and burn-up are being investigated.

Variation of these parameters indicates that the fission products which are released generally fall into one of three categories. These are the volatiles (such as the noble gases and the halogens which are usually released to a larger extent than other fission products), the volatile metals and oxides (such as caesium and ruthenium which show variable releases), and the non-volatiles (such as barium and strontium which are invariably released to a small extent). Further, these studies have shown that three release mechanisms could operate as the result of an accident. The first is release via diffusion through non-molten fuel and subsequent vaporization of fission products, the second is release on fuel oxidation, and the third is release on fuel melting and vaporization. Each of these mechanisms can give rise to significant releases under appropriate conditions. Results indicate, for example, that release from UO_2 via diffusion follows the diffusion laws up to 1 700 °C, and that the release rate accelerates markedly above that temperature, apparently due to grain growth. On the other hand, fission-product release on the air-oxidation of UO_2 is significant only at temperatures below 1 500 °C. This mechanism gives rise to large fractional releases of the rare gases, iodine, tellurium, caesium, and ruthenium. The high release of ruthenium on UO_2 oxidation has been attributed to the formation of the volatile oxide, RuO_4. Fission-product release via air-oxidation would be important in a reactor accident for those portions of the core in which cladding melting temperatures but not fuel melting temperatures were reached; in this case, a substantial fission-product release could occur as the fuel temperatures cooled from 1 500 °C to cooler temperatures. The effect of steam on release *via* the air-oxidation of UO_2 is currently being investigated. The most effective release mechanism is, of course, fuel melting. Here one observes large fractional releases of

many fission products within a few minutes. Fortunately, many of these plateout even on surfaces above 1 000 °C. This was shown in some in-pile melt-down experiments of miniature stainless-steel clad UO_2 fuel pins in the Oak Ridge Research Reactor. These experiments were carried out in flowing helium; the molten period was generally held to five minutes, and resulted in partial vaporization of the fuel. Here over 90% of the ruthenium, tellurium, caesium, and iodine, and over 70% of the strontium, zirconium, barium, cerium, and UO_2 were no longer associated with the original fuel pin. The data, however, indicate that less than 1% of the strontium, zirconium, cerium, barium, and UO_2 were released from the "high-temperature zone" (minimum temperature, 1 000 °C) to cooler portions of the experimental assembly. Large fractions of the more volatile fission products, however, did leave the high-temperature zone. Similar experiments are now being carried out in air and air-stream atmospheres in order to more closely simulate a loss-of-coolant accident.

The physical and chemical forms of the released fission products are being studied both for in-pile and out-of-pile experiments. This type of study is complicated by the interaction of the released fission products with both impurities normally present in the atmosphere, and with reactor-accident debris. These interactions frequently give rise to large fractions of radio-active aerosols as well as anomalous compounds. The behavior of fission-product iodine is particularly notorious in this regard. Under some conditions it tends to react with impurities to form compounds which are not effectively removed by current air cleaning systems. Under other conditions, it appears to adsorb selectively on certain types and sizes of particulate material; both reversible and irreversible adsorption have been observed. This behavior, along with radio-active aerosol agglomeration rate studies, is currently being investigated on a laboratory scale. Further, fission-product transport and deposition are being investigated in a 1 500 ft^3 containment-shell simulant called the Nuclear Safety Pilot Plant. Here experiments will be carried out under a wide range of realistic experimental conditions in order to simulate the different fission-product behavior patterns which would be encountered under different accident conditions. This facility substantially enhances our capability to investigate on a more realistic scale, the transport and deposition of accident-born fission products, and counter measures to control them.

REACTOR CONTAINMENT

Research and development work on reactor containment is devoted to the general problems of defining (*a*) the nature of energy releases which can be expected from credible accidents, (*b*) the ability of containment systems to resist the transient and long-term loads caused by the energy releases, (*c*) the ultimate strength properties of pressure vessels in withstanding rapid pressure loading or missile formation, and (*d*) means of improving existing containment design without adding to the cost or reducing the factor of safety. Theoretical studies and scale-model experiments are an integral part of this program.

On the problem of defining energy releases, loss-of-coolant phenomena for pressurized and boiling water reactors has received considerable attention since the stored thermal energy of such reactor plants is large and can lead to substantial pressure build-up in the containment building. Studies are presently under way to improve analytical models for the prediction of pressure drop and mass flow rate for two-phase flow representing effects during a postulated loss of coolant accident.

A Containment Systems Experiments project has been developed and work is under way in fabrication of experimental facilities at Hanford. Great importance is attached to test data and information to be obtained from this larger scale work, also particularly with respect to reactor location. Existing facilities at Hanford are being utilized in this work. Briefly, the main feature of the project is an outer steel shell of some 30 000 ft^3 designed for 75 psi(g) and a 150 ft^3 pressurized source designed to 2 500 psi(g). A wide range of containment schemes and parameters and so-called "engineered-safeguards" such as sprays, filters, etc., will be studied and tested, as well as the transport of fission products and leakage rate determinations over a wide range of system conditions. Project design and engineering work is well advanced and it is hoped that significant testing will start in the spring or summer of 1965.

Another important project related to containment has been developed, and assigned to the General Electric Company, and the investigative phase of this work is now under way. It deals with the phenomena associated with pipe rupture in pressurized reactor coolant piping. Such failures have been generally postulated as the initial event which subsequently leads to core melt-down and fission product release in water reactors. We believe that better understanding of, and better quantitative information on initiating mechanisms for pipe failure, how defects grow to cracks, how cracks grow to failures, maximum size of rupture, probability of failure, will not only lead to a more rational technical basis for assessing this facet of the nuclear safety problem, but also to preventive design measures, materials and fabrication inspection and other engineering steps to prevent such occurrences.

Other containment related projects which have been completed are mentioned here simply by title since detailed project reports are available. They are

(*a*) investigation by the Stanford Research Institute to determine factors which are important in relation to missile generation and the problems they might pose to containment penetration and therefore containment structure design; (*b*) the distribution of absorption of accident energy for a fast breeder reactor such as the Enrico Fermi Plant and its relation to the so-called reactor top-plug jump problem. This work was done by the US Naval Ordnance Laboratory, and (*c*) work done by Atomics International to determine how normal building structures might be improved for use as low pressure containment for reactors that do not have the stored energy of pressurized water. A handbook on this subject has been published.

RADIATION EMBRITTLEMENT OF STRUCTURAL STEEL

Research and engineering studies on the degradation of structural materials by fast neutrons are sponsored by the Fuels and Materials Branch in our Division of Reactor Development. It is well known that the important property with regard to safety is the change in ductility. The ductile to brittle transition temperature is elevated enough under certain conditions of neutron exposure that the life expectancy of reactor pressure vessels and/or their mode of operation can be seriously affected. At least six different experimental investigations are under way in the US to study the separate and lumped effects of neutron flux intensity, neutron spectrum, specimen temperature, stress during irradiation, the influence of past irradiation annealing, and a few other parameters which represent likely material exposure and operating conditions. Surveillance programs also exist, whereby sample coupons of materials of the same composition as the reactor vessel structure are exposed to the neutron field during the operation of the reactor. Such samples are tested periodically to determine changes in ductility as a function of exposure history. Some tests have also been conducted on materials sectioned from the SL-1 primary vessel structure permitting some observations on the effects of stress during irradiation as well as possible effects of neutron dose rate and neutron energy spectrum variations. The result of the SL-1 materials study shows that the increase in NDT (based on Charpy-V notch data) is only slightly greater (70 °F) than might be predicted by extrapolation of data from accelerated irradiations of similar steels.

One general set of results that seems to be coming out of the various test programs shows that many structural steels, for a given dose level and irradiation temperature, have NDT shifts which can be described by a temperature band 50 to 75 °F wide. Data on NDT shifts from SL-1 and Dresden have tended to confirm the validity of these bands. However, some anomalies exist and must be explained with further experiments. For example, some recent tests on five different steels, including 212B and 17-4 PH, all irradiated equally at the same temperature, have shown a range of NDT shifts greater (up to 150 °F) than the expected band width.

A USAEC report, TID-17682, has recently been published which summarizes the neutron irradiation surveillance practices for 21 different reactor vessels.

A significant contribution to knowledge in this important area should be forthcoming if our plans to carry out various radiation effects tests on a section of the PM-2A reactor pressure vessel materialize. This reactor has been operated by the US Army in Greenland since 1961 and the reactor pressure vessel has been subjected to an integrated neutron exposure of approximately 1×10^{19} nvt.

A comparison between irradiation effects on laboratory type specimens and in an actual vessel should be quite valuable *per se* and at least qualitatively valid for other materials.

FAST REACTOR SAFETY STUDIES

The major emphasis of fast reactor safety studies has been on fuel melt-down phenomena. Since fast reactors contain sufficient fuel for several critical masses arranged in a non-optimum geometry, it is not difficult to hypothesize a class of accidents where fuel melt-down is followed by reassembly into a more reactive configuration. This can lead to a super-prompt excursion resulting in extremely large energy release. The analysis of such accidents is hampered by the lack of knowledge of the mechanisms affecting fuel element failure and movement of materials as well as their effect on reactivity.

In-pile fuel melt-down experiments are being conducted in the Transient Reactor Test (TREAT) facility to investigate the complex factors influencing fuel failure and movement. The aim of the program, conducted by Argonne National Laboratory, is to provide experimental information together with theoretical analyses needed to assess the consequences of potential melt-down accidents in specific fast reactor designs.

In the softer neutron spectra of large ceramic cores both the Doppler effect and the sodium void coefficient can become significant. The Doppler broadening of nuclear resonances can possibly terminate a fast reactor excursion before core damage occurs or at least limit the extent of damage. The sodium void coefficient, on the other hand, may be at least locally positive in large fast ceramic reactors. Both of the above effects have been under extensive analytical investigation by various groups. Several programs are under way using critical facilities, ZPR (Zero Power Reactor) III and VI at Argonne National Laboratory and the Epithermal Critical Experiments Laboratory (ECEL) at Atomics

International, to measure the temperature dependence of resonance absorption for various fissile and fertile materials in a number of fast reactor spectra. Sodium reactivity worths, including sodium void effects, will also be measured in these facilities. A number of studies are under way to investigate sodium boiling and the dynamics of the expulsion of sodium from reactor coolant channels. Knowledge of void distributions under abnormal operating conditions is necessary for the assessment of fuel failure as well as for the prediction of the magnitude of the reactivity effect associated with sodium voids. Furthermore, both the Fast Reactor Test (FARET) and Southwest Fast Oxide Reactor (SEFOR) projects will to a great extent be devoted to investigations related to the safety of fast reactors.

It seems fairly clear that in the future, as the development of fast breeder reactors increases in tempo, that part of our nuclear safety program will correspondingly increase.

ENGINEERING SCALE SAFETY TESTS

In reactor safety research, the basic approach to an understanding of complex excursion and other accident phenomena is to separate variables and to study one effect at a time, whenever possible. From such investigations and analyses one then attempts to synthesize or predict a total real event. This leads to some analytical difficulties for the reactor safety analyst or evaluator when he tries to extrapolate data from laboratory scale experiments to the large scale and complex situations prevailing during an assumed reactor accident. He is also faced with the difficulty of assuring himself that his synthesis is complete, i.e., that he has delineated properly and analyzed sufficiently all the important phenomena and their inter-actions. It is therefore believed necessary to conduct engineering scale tests of reactor accident situations to (*a*) determine the degree of validity which exists in the various assumptions used for calculating accident consequences, in other words that the important phenomena have been identified and are understood; and (*b*) demonstrate the functioning and reliability of engineered safeguards designed to prevent an accident and/or to control its consequences.

Accordingly, plans have been fairly well advanced for carrying out such tests. Initially these tests would be directed to two principal types of potential water reactor accidents; the loss-of-coolant accident which in the case of water reactors can lead to fuel meltdown and subsequent release of fission products to the reactor containment and possibly to the environment; and the nuclear excursion which can lead to core damage or initiate a loss-of-coolant accident. This initial emphasis on water cooled and moderated power reactor systems is a logical consequence of the prominence of these systems in the developing US nuclear power program over the next ten to fifteen years.

New facilities are being developed or planned for these tests. The major one and the one most advanced is designated LOFT (Loss of Fluid Tests). The major features of this installation are a railroad dolly-mounted 50 MW (th) pressurized water reactor and a full scale containment shell which can receive the dolly-mounted reactor for the test and from which the reactor may be rail transported to a large hot shop for post-test examination.

Prior to the actual loss of fluid test a number of tests related to containment integrity and behavior will be done. There is also planned a series of blowdown tests, using a dummy core, to investigate various thermal and mechanical or physical phenomena associated with a sudden loss of water from different size main coolant piping ruptures at different locations, etc.

Engineering design for this major undertaking is well along. Construction has been started and the actual test program noted above is scheduled to begin in 1966 with the actual reactor loss-of-fluid test scheduled in 1967.

All of the land based engineering test program or STEP (Safety Test Engineering Program) is being conducted by the Phillips Petroleum Company at the US National Reactor Testing Station in Idaho. The LOFT facility is being engineered by Kaiser Engineers with the support of Babcock and Wilcox. Other parts of the nuclear safety program are also contributing direct support to the LOFT project.

RADIOACTIVE WASTE MANAGEMENT

The satisfactory management of radioactive waste from the nuclear energy industry continues to be an essential part of the over-all AEC nuclear safety program. Significant advances in waste processing technology have been achieved which further reduce the volumes and quantities of radioactive material being discharged to our environment.

As the industry continues to grow, the nature of the waste management problem can be delineated as (*a*) the disposal or concentration of large volume low-activity waste, and (*b*) the long-term, or permanent storage or disposal, of concentrated high-level waste. When nuclear activities were small in scale, wastes with very low specific activities were discharged to the environment without significantly raising the radiation background level. Controlled environmental discharges are becoming more limited, not only due to the expanding industry, but also because acceptable environmental limits are being continuously reduced. Therefore, it has been necessary for the waste management research and development program to develop improved and more

efficient methods for treating and concentrating large volumes of low-activity wastes, and to provide disposal systems for these waste concentrates. Integrated with this program are comprehensive environmental investigations to: (*a*) determine the ultimate fate of specific radionuclides in land, in water, and in air environments, and (*b*) establish reasonable technical criteria for safe disposal of low-level radioactive effluents into the environment.

A prime example of environmental studies which are associated with waste management operations is the comprehensive stream investigation being carried out on the Clinch and Tennessee Rivers, below Oak Ridge National Laboratory. The principal objectives of the program are to determine and understand the fate and distribution of specific radionuclides (90 Sr, 137 Cs, 106 Ru and 60 Co) as discharged to the Clinch River from Laboratory operations. It is significant to note that most of the ORNL radioactive waste entering the Clinch River passes through the river system in the water phase with bottom sediments and the biomass being of minor import from a health and safety standpoint.

Recent studies at Oak Ridge on low activity waste treatment have indicated that a phenolic resin ion-exchange process has provided treatment efficiencies of 99.9% for strontium and caesium, with volume reductions of approximately 2 000. The disposal of intermediate level wastes by hydraulic fracturing of shale or other suitable geologic formations (a technique which has been obtained from the petroleum industry) has been under extensive development at ORNL during the past four years. The method consists of injecting a waste-cement-clay mixture under high pressure into an impermeable formation by fracturing. A full-scale engineering pilot plant has been constructed at Oak Ridge and demonstration tests with actual intermediate level waste are now being carried out.

The program for developing treatment technology to convert high-level liquid waste to an inert solid form and to provide for safe ultimate storage of these wastes in selected geologic formations, has received major emphasis in the US during the past several years. The largest development effort in this area has been the installation of a 230 l/h Waste Calcination Facility at the Idaho Chemical Processing Plant for the fluidized bed calcining of acid aluminum nitrate waste from MTR-type fuels. "Hot" operation of this plant commenced in December 1963, and satisfactory, full-scale operation of the calciner has been carried out since that time in an effort to reduce the volume of waste now in storage at the ICPP. The power reactor waste solidification program has now reached the stage where a 40 l/h Waste Solidification Engineering Prototype Plant is now under construction at the Hanford Laboratories. Present plans call for demonstration of the pot and radiant spray calcination processes and/or modifications of these techniques, which will result in a more stable ceramic end product, as well as a continuous phosphate glass process during the latter part of 1965.

In conjunction with the high-level waste solidification program is the investigation of selected geologic formations for the storage or ultimate disposal of these solid wastes. Salt formations have been chosen as the most optimum disposal media because of their unique geologic characteristics. Extensive laboratory and field studies in salt have resulted in the design and construction of a field experiment, using short-cooled ETR fuel elements which will reproduce the thermal and radiation characteristics of full-scale power reactor waste such as would exist in a pot containing calcined solids. Operation of this field demonstration is planned for early 1965.

NUCLEAR SAFETY ANALYSIS AND EVALUATION

The activities in the area of nuclear safety analysis and evaluation are only partially of a research and development nature. From an administrative standpoint it covers internal co-ordination of the various nuclear safety programs in the AEC and external co-ordination with the regulatory and control groups in the agency. It also covers the nuclear safety information program.

Of a more programmatic nature is the work of investigation of reactor operating experience applicable to nuclear safety. Test and research reactors were first studied in this regard by the Holmes & Narver Company and the work is now being done on large scale power reactors. Analysis and evaluation of actual operating experience should provide valuable insights of safety significance. Other investigations related to pressure vessel design, including development of codes for nuclear vessels, and the application of probabilist techniques and methods to reactor safety evaluation are other examples of work in this area.

DISSEMINATION OF NUCLEAR SAFETY INFORMATION

Because of the diversity and scope of research and development programs related to the development of reactor materials, components and systems, those who must design safeguards and evaluate safety are faced continually with keeping abreast of current developments. This requirement presents a particularly challenging problem since unlike the engineer or scientist who concentrates on a specific problem or specialized field, those concerned with the broad problem represented by nuclear plant safety are faced with developments in a wide spectrum of activities. It is therefore important that some mechanism be established

for extracting the safety essence from a wide variety of reports to make this information readily available to those who must use the information in reactor design or safety assessment. Several ways have been established in the United States to collect, digest and disseminate technical information on a wide variety of reactor safety oriented programs. Of particular interest is the establishment by the USAEC in July 1963 of a Nuclear Safety Information Center at the Oak Ridge National Laboratory. The Center collects, evaluates, and disseminates safety information generated in the US and abroad. It is staffed by a group of nuclear experts who will provide personalized assistance, as time allows, on safety problems of interest to the nuclear community. All the modern concepts of systems of information retrieval and dissemination using computers are being utilized in this center and it is believed that its work will result in major contributions to the field.

ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/282 Etats-Unis d'Amérique

Programme de recherches et de développement en matière de sûreté nucléaire aux Etats-Unis

par J. A. Lieberman

Les objectifs du programme de recherches et de développement en matière de sûreté nucléaire de la Commission de l'énergie atomique des Etats-Unis sont de fournir une base technique solide pour la promotion rationnelle de la sûreté nucléaire dans la mise au point, l'implantation, l'étude, la construction et l'utilisation de réacteurs et installations annexes dans les applications terrestres, spatiales et marines. Un programme étendu d'essais analytiques, expérimentaux et techniques est en cours de réalisation pour obtenir des informations dans différents domaines techniques. Ces domaines comprennent: cinétique des réacteurs, réactions combustible-réfrigérant, échappement de produits de fission, enveloppe de sécurité des réacteurs, sûreté des réacteurs rapides; essais pratiques de systèmes nucléaires terrestres et spatiaux, protection et rétablissement à la suite d'accidents; études sur le traitement, le stockage et la destruction des déchets radioactifs; analyse et évaluation de projets dans le cadre des objectifs du programme. Ce programme est complété par certaines études sur la sécurité entreprises au titre de différents projets de réacteur particuliers.

Une partie importante du programme est applicable à la mise au point de réacteurs pour centrales nucléaires civiles et à la solution des problèmes d'implantation en accord avec les règles de sécurité approuvées par l'AEC. Les activités liées à l'utilisation croissante d'engins nucléaires dans l'espace sont divisées en recherche et développement, essais au sol et essais en vol. Elles comprennent des lancers réels de véhicules spatiaux portant des équipements spécialement étudiés, et des essais de destruction au sol de réacteurs spatiaux pour étudier directement le comportement du réacteur et les conséquences d'un accident.

Le programme SPERT (Special Power Excursion Reactor Tests) a fourni des renseignements très étendus sur le comportement de réacteurs refroidis et modérés à l'eau légère lors de conditions transitoires, y compris des états transitoires destructifs dans des cœurs du type en plaques métalliques ou en barreaux de UO_2. Des études sur la fusion du combustible effectuées à l'installation TREAT fournissent toujours plus de renseignements sur les phénomènes connexes en matière de cinétique et de libération de produits de fission. La portée de ces études sera bientôt étendue grâce à la future installation pour l'étude de ruptures en puissance, qui permettra d'essayer de grands ensembles de combustible par destruction. Une installation d'essais de 50 MW(th) est projetée et permettra d'effectuer des essais de fuites de fluide à l'échelle réelle dans un réacteur de puissance. Ces essais, qui seront les premiers du genre, seront effectués au Centre national d'essais de réacteurs dans l'Idaho. Des études sur le traitement des déchets hautement radioactifs ont atteint le stade où des usines pilotes appliquant des techniques de conversion des déchets liquides en solides susceptibles d'être stockés de manière permanente fonctionnent avec des résultats satisfaisants.

Au cours des deux dernières années, une attention toute particulière a été portée à l'amélioration de l'échange d'informations sur l'expérience acquise en vue de leur application à l'étude et la réalisation du programme d'ensemble. Cela a abouti à la création du Centre d'information sur la sûreté nucléaire d'Oak Ridge. On a établi des méthodes pour l'échange ininterrompu d'informations avec les secteurs industriels privés et des firmes industrielles particulières effectuant leurs propres études à l'échelle réduite sur la sûreté de certains types de réacteurs. Des mesures sont prises pour améliorer encore les échanges d'informations et pour obtenir une participation plus intime de l'industrie au programme de l'AEC en matière de sûreté.

A/282 США

Программа исследований и опытно-конструкторских работ по ядерной безопасности в США

Дж. А. Либерман

Отдел исследований и опытно-конструкторских работ по ядерной безопасности при Комиссии по атомной энергии США стремится создать надежную технологическую базу для внедрения принципов ядерной безопасности в разработку, выбор местоположения, проектирование, строительство и эксплуатацию каких бы то ни было ядерных систем и связанного с ними вспомогательного оборудования на суше, на море и в космосе. В связи с этим ведутся обширные исследования аналитического, экспериментального и технического характера в разных областях ядерной технологии.

Эти области включают: кинетику реакторов; реакции при охлаждении топлива; утечку продуктов деления; оболочки реактора; вопросы безопасности, связанные с работой реакторов на быстрых нейтронах; практические испытания ядерных систем как на земле, так и в космосе; мероприятия, связанные с ликвидацией последствий аварии; исследование методов обработки радиоактивных отходов, их хранение и удаление; анализ и оценку всех существующих установок с точки зрения целей, преследуемых настоящей программой.

Эта программа дополняет и развивает исследования различных вопросов безопасности, которые ведутся на отдельных ядерных установках.

Большая часть работы Отдела исследований и опытно-конструкторских работ по ядерной безопасности связана с разработкой энергетических реакторов для гражданских целей, а также с вопросом выбора их местоположения, которое должно соответствовать принципам безопасности, установленным Комиссией по атомной энергии США.

Деятельность, связанная с развитием применения ядерных устройств в космосе, делится на три части: общие исследования и разработки, испытания на земле и летные испытания. Она включает осуществление запуска космических ракет со специально рассчитанной нагрузкой, а также наземные испытания реакторов для космических полетов на разрушение, помогающие точно определить последствия возможных аварий и поведение реакторов в таких случаях.

Программа испытаний, получившая название SPERT (опытный реактор для испытаний на выбросы мощности) позволила собрать большое количество материалов о переходных режимах в реакторах, охлаждаемых и замедляемых обычной водой, включая переходные режимы, сопровождающиеся разрушениями в активных зонах с металлическими пластинками или со стержнями из окиси урана.

Опыты с расплавлением топлива, проведенные на установке TREAT (опытный реактор для изучения переходных процессов), намного расширили область наших знаний о кинетических явлениях процесса и выделении продуктов деления.

Сфера этих исследований намного расширится в связи с предстоящей постройкой специальной установки для изучения энергетических вспышек, на которой можно будет проводить испытания на разрушение больших топливных сборок.

В настоящее время проектируется специальная экспериментальная установка мощностью 50 *Мвт* (тепл.), которая даст возможность проводить испытания энергетических реакторов в условиях полной потери теплоносителя.

Этот опыт — первый в своем роде — будет проведен на Национальной станции по испытанию реакторов в штате Айдахо.

Исследование методов обработки высокоактивных отходов уже достигло уровня, при котором оказалось возможным создать комплекс полузаводского масштаба, в котором жидкие отходы перерабатываются в твердые, после чего окончательно удаляются. Эта установка работает вполне удовлетворительно.

За последние два года особенно большое внимание уделялось вопросу улучшения обмена информацией, относящейся к планированию и выполнению совокупности всей программы. В результате этого в Ок-Ридже был создан Информационный центр по ядерной безопасности.

Установлена связь для постоянного обмена новейшей информацией как с частными предприятиями, так и с правительственными учреждениями, ведающими установлением норм и правил для ядерной индустрии в целом. Такая же связь установлена с теми немногими промышленными фирмами, которые также ведут исследования в области ядерной безопасности в той мере, как это относится к реакторам какого-либо конкретного типа.

Предпринимаются также шаги к дальнейшему улучшению обмена информацией и более широкому участию частных фирм в программе безопасности, намеченной Комиссией по атомной энергии США.

A/282 Estados Unidos de América

El programa de investigación y desarrollo de seguridad nuclear en los Estados Unidos

por J. A. Lieberman *et al.*

El objetivo del programa de investigación y desarrollo de seguridad nuclear de la Comisión de Energía Atómica de los Estados Unidos es suministrar cono-

cimientos tecnológicos que sirvan de base para el fomento de la seguridad nuclear en el desarrollo, la localización, el diseño, la construcción y operación de tipos de reactores y otras instalaciones relacionadas con ellos en sus aplicaciones terrestres, espaciales o marítimas. Se está desenvolviendo un amplio programa de investigaciones a escala, analíticas, experimentales y en el campo de la ingeniería, para obtener información en varios campos técnicos. Estos incluyen la cinética de los reactores, reacciones del refrigerante de combustible, liberación de los productos de fisión, confinamiento de los reactores, seguridad de los reactores rápidos; pruebas de ingeniería de sistema nucleares, terrestres y espaciales; protección y recuperación después de un accidente; investigaciones sobre la manipulación, almacenamiento y distribución final de los desechos radiactivos; y el análisis y evaluación de proyectos en su relación con los objetivos del programa. Este programa apoya y complementa investigaciones sobre la seguridad, que se llevan a cabo bajo proyectos específicos relacionados con los reactores.

Una gran parte del programa de investigación y desarrollo de seguridad nuclear puede aplicarse al desenvolvimiento de reactores de energía para usos civiles y a la solución de problemas de localización, de acuerdo con las regulaciones de seguridad aprobadas por la Comisión de Energía Atómica. Las actividades relacionadas con el uso creciente de dispositivos nucleares en el espacio están divididas en actividades de investigación general y desarrollo, pruebas terrestres, y pruebas de vuelo. Estas pruebas comprenden lanzamientos de vehículos espaciales con cargas de prueba especialmente diseñadas y pruebas terrestres de reactores espaciales en las que se provoca la destrucción de los reactores para determinar el comportamiento del reactor y las consecuencias de un accidente.

El programa SPERT (Special Power Excursion Reactor Tests) ha suministrado una gran cantidad de información básica acerca del comportamiento transitorio de tipos de reactores refrigerados y moderados por agua ligera incluyendo estados transitorios destructivos en núcleos metálicos laminados y en núcleos de tipo de barra de UO_2. Los estudios sobre la fusión del combustible llevados a cabo en la instalación TREAT están suministrando conocimientos en cantidad creciente acerca de la cinética y el desprendimiento de productos de la fisión relacionados con la dicha fusión. El alcance de estos estudios será extendido con la construcción de la instalación de Power Burst, que será capaz de provocar y estudiar el proceso de destrucción de grandes conjuntos de combustible. Se está diseñando una instalación de pruebas de ingeniería de 50 MW(t) la cual hará posible, en escala natural, la prueba de pérdida de fluido en un reactor de potencia. Esta prueba, que será la primera de su clase, será hecha en la Estación Nacional de Pruebas de Reactores de Idaho. Las investigaciones acerca del tratamiento y liberación de desechos altamento radiactivos están tan adelantadas que dan resultados satisfactorios las instalaciones experimentales que incorporan los procedimientos para la conversión de desechos líquidos en sólidos convenientes para su eliminación permanente.

Se ha prestado especial atención en los dos últimos años al mejoramiento del suministro de información experimental para el planeamiento y ejecución del programa general. Esto ha originado la creación de un Centro de información sobre la seguridad nuclear en Oak Ridge. Varios mecanismos han sido establecidos y usados para proveer un intercambio continuo de información con los sectores gubernamentales, con los sectores de la industria privada y con firmas industriales concretas que realizan investigaciones limitadas sobre la seguridad de tipos particulares de reactores. Se están tomando las medidas necesarias para lograr el mejoramiento del intercambio de información y una mayor participación de la industria en el programa de seguridad de la Comisión de Energía Atómica.

Reactor excursion behavior *

By W. E. Nyer,** G. O. Bright ** and R. J. McWhorter ***

Information accumulated in recent years on reactor excursion behavior indicates that significant sectors of the reactivity accident problem are well understood. Of particular practical importance is the high degree of safety that can be achieved by designing self-protective properties into both the inherent neutronic characteristics and the external controls of reactors systems. In many instances, the reactivity accident in power reactor systems can, through proper design, be reduced to secondary importance in the consideration of the maximum credible accident. The data which form the basis of this atmosphere of confidence will be reviewed, and an example of the interaction of design considerations with possible reactivity accidents will be given.

EXCURSION DATA

Available power burst data have covered a wide range of conditions and provide considerable information for attacking many excursion sefety problems. The data will be grouped according to the three aspects of the reactivity accident on which the studies have focused: understanding the kinetics of excursions by identifying (or verifying) the various inherent reactivity effects and by developing appropriate analytical descriptions; investigating the effects of power plant type variables upon the fundamental kinetics; and investigating the nature and consequences of destructive reactivity accidents representative of the largest-scale accidents of this type. From the standpoint of practical safety applications, the data also naturally group into three categories distinguished by the extent of physical damage incurred in the tests. These groups can be characterized by broad reactivity ranges: low (no physical damage), medium (moderate damage), and high (destruction). In these ranges, different shut-down mechanisms dominate.

Studies of reactor kinetics

This grouping includes the work carried out with the systems listed in Table 1, which describes briefly the reactors and some of the principal factors in the experimental and analytical studies. Excursion behavior for these different types of systems has been remarkably similar. For example, the curves of peak power attained in self-limiting bursts as a function of reciprocal period, (α), are qualitatively alike, and for thermal reactors, fall into a narrow range [14]. Many features of excursion behavior, such as peak power, energy release, burst shape, etc., can be described by simple analytical models using space-independent kinetics and an energy-dependent reactivity coefficient [15]. For reactors with prompt, essentially single-mode shut-down mechanisms, such as TREAT, TRIGA and the SPERT oxide core, these reactivity coefficients may be calculated directly and compare well with the experimentally determined coefficients.

For TRIGA, detailed analyses [16, 17] also showed that the fundamental spatial mode dominates during

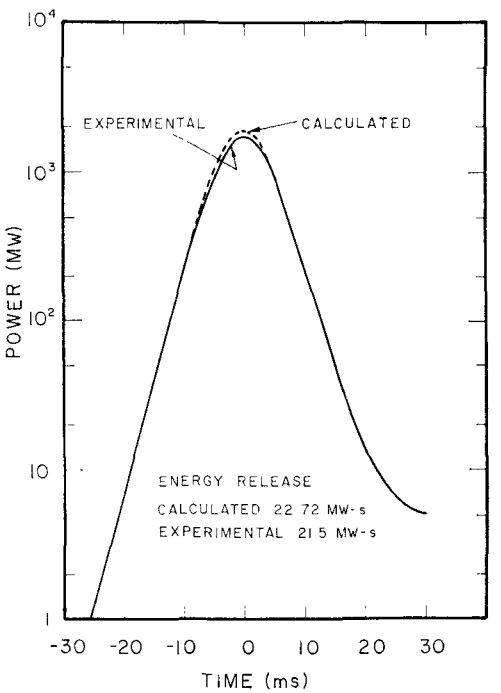


Figure 1. Comparison of calculated and experimental values for TRIGA reactor

* With contributions by R. Scalettar (General Atomic, San Diego, California), A. A. Jarrett (Atomics International, Canoga Park, California) and W. R. Stratton (Los Alamos Scientific Laboratory, Los Alamos, New Mexico).

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Table 1. Reactor excursion studies

Reactor	Type	$\alpha_r(s^{-1})$	Range of reactivity studies		Principal shut-down mechanisms
			\$(prompt)	$\alpha(s^{-1})$	
TRIGA [1]	Heterogeneous, H_2O-ZrH-moderated, U-ZrH fuelled	100	0.74–3	0.57–345	Prompt leakage increase due to excitation of bound hydrogen in ZrH
TREAT [2]	Homogeneous, graphite-moderated, UC fuelled	8	$\leq$ 4.15	$\leq$ 25	Prompt leakage increase with moderator heating
KEWB [3]	Homogeneous, H_2O-moderated, uranyl sulphate fuel	127	$\leq$ 5.8	$\leq$ 900	Prompt moderator expansion, void formation
Fast core [4,5] (Godiva, FBR)	Bare, enriched-uranium metal, fast	1×10^6–5.5×10^5	$\leq$ 1.11	$\leq 8.6 \times 10^4$	Prompt fuel expansion
SPERT	Plate-type, H_2O-moderated [6, 7, 8], fully-enriched, U-Al fuel	94–140	0.35–3.55	$\leq$ 320	Prompt fuel expansion, delayed moderator expansion and steam void formation
	Plate-type, D_2O-moderated [9], fully-enriched, U-Al fuel	10	0.40–3.00	$\leq$ 20	Prompt fuel expansion, delayed moderator expansion and steam void formation
	Plate-type, H_2O-moderated [10, 11, 12], fully-enriched, UO_2-SS fuel	310–435	0.40–2.00	$\leq$ 300	Prompt fuel expansion, delayed moderator expansion and steam void formation
	Rod-type [13], H_2O-moderated, low-enrichment UO_2 fuel, SS clad	280	0.50–2.60	$\leq$ 450	Prompt Doppler effect, delayed moderator expansion and steam void formation

a pulse, the admixture of higher modes is small (and calculable), and the temperature coefficient and lifetime may be obtained from modified perturbation theoretical formulae, with an appropriate weighting associated with local temperature. Figure 1 shows the agreement of calculated results with experiment.

The SPERT oxide core transient experiments demonstrated the effectiveness of the Doppler mode of self-shut-down and provide a basis for analysis of accidents in similar power reactor systems. In the analysis of these excursions, a simple calculational model was developed for the temperature-dependent Doppler effect in a thermal oxide core with a non-uniform temperature distribution [18]. An analytical solution of the prompt-approximation, space-independent neutron kinetic equation was obtained using Doppler feedback as the only shut-down mechanism. Corrections were applied for the small contributions from moderator heating. The form of the Doppler temperature-dependence has been predicted to vary as the square root, the logarithm, or the inverse square root of the temperature. The use of a square-root dependence with the simple model produced a systematic agreement between calculated and experimental effects over the entire range of adiabatic fuel temperature rises in the short-period SPERT tests, whereas the other dependences gave results which differed significantly from experimental results. This strongly implies the validity of the square-root temperature dependence for the Doppler effect in a thermal oxide core.

The state-of-the-art is not as well advanced in describing the behavior of a system which has a complex, perhaps delayed, shut-down process involving heat transport and phase changes such as in the SPERT plate-type cores [19]. For example, the error in calculating the reactivity effects for these cores is several times as great as that for the oxide core. A question could exist as to whether the source of the inability to better match theory and experiment lies in the understanding of the physical shut-down effects or in the simple way of handling the neutron kinetics. From the success of calculations for systems wherein the shut-down processes are dominated by a single and relatively simple mechanism, the problem would appear to lie in the treatment of the mechanisms of shut-down rather than in the way in which the neutronic behavior is handled.

Since, in practical considerations many dollars of excess reactivity may be potentially available for excursion initiation, and the damage resulting from excursions is influenced by the localized energy-density which can be achieved, it is of interest to compare the excursion data shown in Figures 2 and 3. Figure 2 shows the average energy density, on a volume basis, which is achieved by various experimental reactors as the initial excess reactivity is increased. Potential damage, of course, must be evaluated on the

basis of the properties of the material in which the fuel is contained. Figure 3 shows the derived energy-density shut-down coefficient. The figures point out three salient features: the high shut-down efficiency of reactors with large, prompt shut-down mechanisms, such as the prompt temperature coefficient that TREAT and KEWB possess; the small differences in shut-down efficiency for reactors not possessing this feature; and that many thermal reactors can be designed to self-control the addition of large amounts of excess reactivity without serious damage.

From the broad point of view of the safety evaluation of power reactors, the over-all significance of this work goes beyond the demonstrated ability to understand and predict detailed behavior of individual systems. The principal result of this group of experiments, by their very scope and depth, is a less tangible but more important effect. The determination of the existence and verification of the effectiveness of a great variety of quenching mechanisms, and the demonstration that the magnitude of these inherent shut-down mechanisms can be great enough to provide protection against even large reactivity perturbations, lead to confidence with regard to the state of understanding of the excursion problem and confidence that self-protection can be designed into these systems.

Studies of operational variables

In operating power plants, the conditions under which excursions might occur, and the consequences that might follow, are influenced by the variables of plant operation, e.g., (in a boiling or pressurized-water reactor) the steady-state pressure, coolant flow rate, temperature, power, and even the control-rod pattern (which may determine the effective core size). To date, only the SPERT plate-type reactors have been used in investigations of these factors, and the results are not in all cases applicable to systems with long heat-transfer time constants. The effect on the kinetic response of a reactor to changes in these system parameters is a complex situation that is not amenable to a concise, simple and unambiguous presentation, but the general trends can be discussed in a qualitative fashion.

Increasing system temperature (ambient to 400 °F) [20, 21]

In general, increasing the system temperature at constant pressure results in a reduction of peak power, Φ_m; energy to peak power, E_m; fuel plate surface temperature rise at peak power, $\Delta\theta_m$; and maximum fuel plate surface temperature rise, $\Delta\theta_{(max)}$ for all initial periods. The reduction is largest for the shorter-period tests and for temperatures that are nearest to

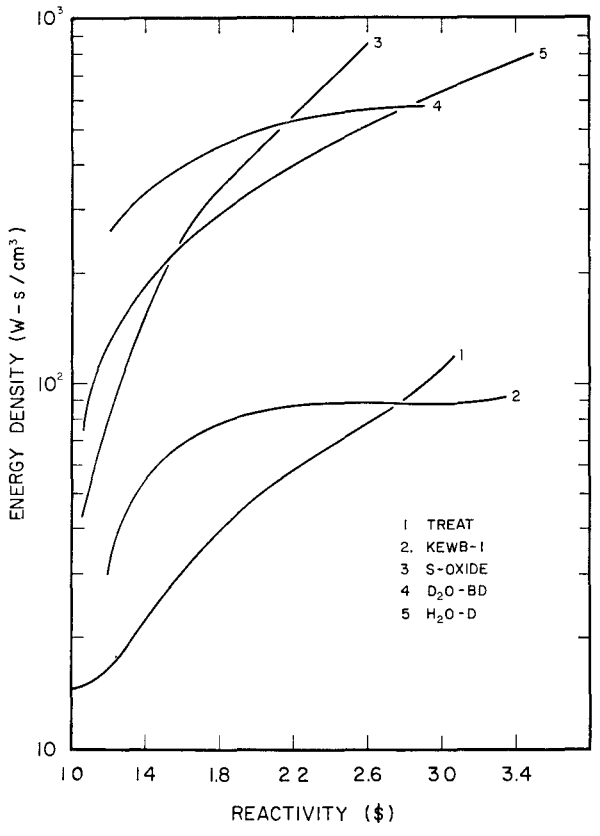


Figure 2. Energy density at time of peak power vs. initial reactivity insertion

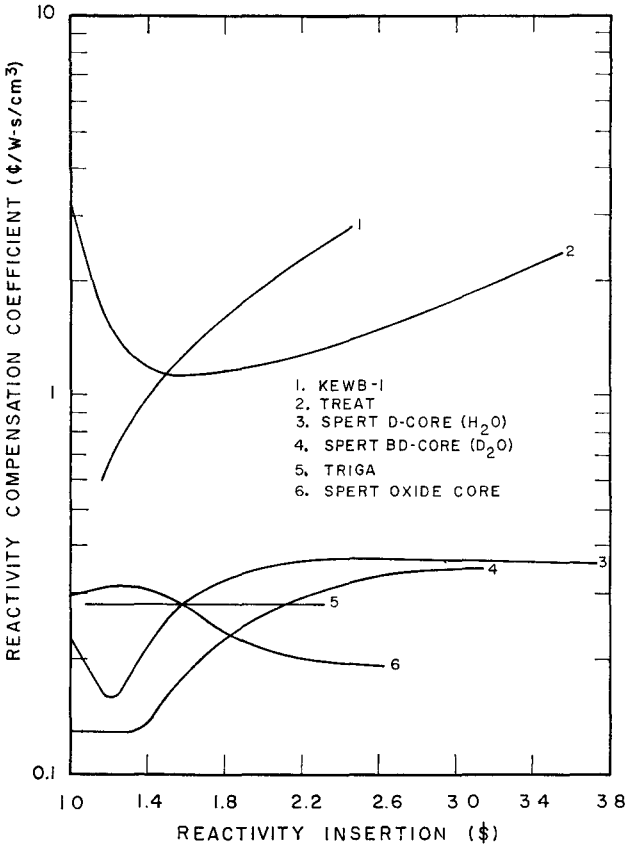


Figure 3. Energy density reactivity compensation coefficient vs. initial reactivity insertion

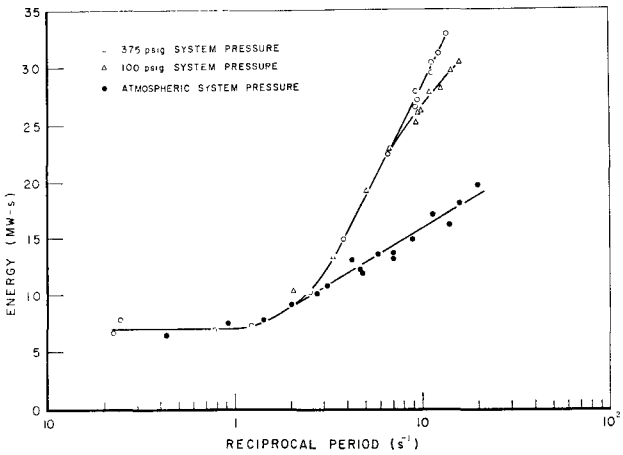


Figure 4. Energy release vs. reciprocal period for various system subcoolings — SPERT 2

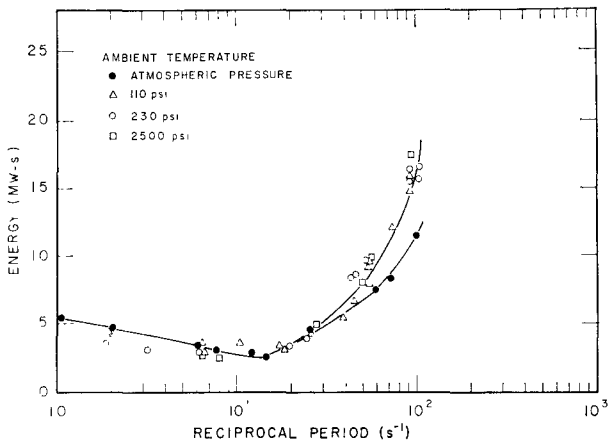


Figure 5. Energy release vs. reciprocal period for various system subcoolings — SPERT 3

the saturation temperature. The actual fuel plate surface temperature (initial temperature plus rise) for the peak power and maximum conditions is increased at the longer periods and decreased at the shorter periods.

Decreasing system (moderator) density (approximately 14%) [22, 23]

These tests were performed by increasing the system temperature and pressure and maintaining constant subcooling. The results were generally of the same type as observed for the increase in system temperature and were consistent with those expected from the increase in the expansion coefficient of water as the density decreases.

Decreasing system subcooling (600 °F to 0 °) (decreasing system pressure 2 500–0 psig) [23, 24]

A decrease in subcooling has no observed effect on longer-period tests, where boiling does not contribute to self-shut-down. For shorter-period tests, where boiling does normally contribute to shut-down, a decrease in subcooling resulted in a decrease in Φ_m, E_m, $\Delta\theta_m$, θ_m, $\Delta\theta_{(max)}$, and $\theta_{(max)}$. The effect is larger as the period of the test is decreased. The major effect of pressure change occurs in the 100–0 psig range, which is consistent with the change in saturation temperature with pressure. Figures 4 and 5 show this effect for SPERT 2, where boiling is the principal shut-down mode for α greater than ~ 3, and for SPERT 3, where boiling is of secondary importance in shut-down.

Increasing system flow (0 to 18 ft/s) [21, 23, 25]

For long periods ($>$ 100 ms), the addition of forced coolant circulation tends to eliminate the initial power-peaking characteristics of the typical power excursion. The power instead rises monotonically until equilibrium is established between heat production and removal. For intermediate periods (100 ms to 50 ms), the power rises to slightly higher initial peaks than for tests without flow, and the power burst shape tends to be slightly broadened. For short periods ($<$ 50 ms), the addition of forced coolant circulation has no significant effect on the power burst as the coolant transit time through the core, ($\sim$ 170 ms), is long compared with the period of the burst. As would be expected from the power behavior, the tests with forced coolant circulation resulted in higher total energy releases than for the corresponding tests without flow. In the short-period region where there was little effect on the peak power or burst shape, there was little difference in the energy released during the burst.

Although the reactor powers that were attained with flow were, in all cases, equal to or greater than those without flow, the fuel-plate surface temperatures were observed to decrease with increasing flow.

Changes in initial power (0–100 kW) [26, 27]

Some testing has been performed with initial power as a variable. For equivalent reactivity insertions, an increase in initial power results in (*a*) a longer-period excursion, and (*b*) reduced peak power. Over the range of the tests, the resulting effects were very weakly dependent upon starting power. However, tests have yet to be conducted under power conditions typical of operating power reactors.

Transient pressures were not of significance from a safety point of view. Indeed, transient pressures sufficiently large to be a safety problem probably can be generated only as a consequence of fuel melt-down or vaporization.

In summary, the experiments indicate that although the effects on reactor kinetic behavior of the operating plant parameters can be complex, the effects are qualitatively understood. Most important, the capability of these plants to safely self-limit excursions induced by reactivity insertions considerably in excess of prompt critical was not significantly changed by operation under power-plant-type conditions.

Tests in the destructive range

The extension of excursion tests into the potentially destructive region has been desirable for a number of reasons. One very important one is the concern that some major destructive effects might not be extrapolable from nondestructive data, that is, that their onset might in effect be a threshold phenomenon. Thus, there is a need for determination of the conditions necessary to produce a destructive burst. Furthermore, uncertainties in many fundamental quantities under these extreme conditions makes preanalysis of the consequences difficult. Finally, there is the desire to demonstrate that nothing fundamental has been overlooked in the consideration of major postulated accidents.

Table 2 summarizes the main features of the tests carried out to date, with the available results of the SL-1 accident included for completeness. BORAX 1, SL-1, and the SPERT 1 D-core were highly-enriched uranium fuelled, aluminum-plate cores with somewhat differing physical characteristics; the SPERT 1 oxide core was made up of low-enrichment UO_2-fuelled stainless-steel clad rods; and the SNAPTRAN 3 test was of a ZrH-U fuelled reactor designed for space applications. BORAX 1, SL-1 and SNAPTRAN 3 were essentially single destructive excursions, but the SPERT 1 D-core tests were a series which progressed through limited melting of the core to the final destructive excursion. The oxide core tests, although damage to the core was experienced, did not result in major destructive effects. The operability of the reactor facility was not impaired.

The tests of greatest interest are the SPERT D-core and the SPERT 1 oxide core. The D-core results indicate clearly the existence of a destructive threshold effect with increasing reactivity. This is shown by the fact that for two preceding tests, in which $\sim 0.3\%$ and $\sim 3.0\%$ of the core were melted, no pressure effect beyond that normally extrapolable from previous tests was observed. The final test, on the other hand (Figure 6), in which $\sim 35\%$ of the core melted, revealed a very sharp, large pressure pulse which occurred after the power burst had been terminated

Table 2. Destructive power excursion summary

Reactor	Reactivity addition ($)	α(s^{-1})	Peak power (MW)	Energy release (MW s)	Maximum temperature (°C)	Maximum energy density (watt seconds/cm^3)	Maximum pressure (psi)	Remarks
BORAX 1 [28]	3.1	384	$\leq$ 19 000	135	$\leq$ 1 800	$\leq$ 6 500	6 000–10 000	Destroyed core, vessel, and some associated equipment. Small fission-product release. Steam explosion proposed as cause.
SL-1 [29]	3.0	280	$\sim$ 19 000	133	$>$ 2 075	$>$ 7 300	10 000	Destroyed core, bulged vessel, local fission-producet contamination. 10% fission-product release. Steam explosion—minor contribution from metal-H_2O reaction
SPERT 1 [30]	2.6	200	1 130	11	585	2 000	7	Melted $>$ 0.5% of core
D-12/25	2.7	218	1 270	19	680	2 300	8	Melted $\sim$ 3% of core
	3.55	313	2 250	31	1 360	4 600	$\leq$ 4 000	Melted $\sim$ 35% of core. Destroyed core and associated equipment, bulged tank. $\sim$ 4% fission-product release. Probable steam explosion—Al_2O_3 analysis indicates $\sim$ 3.5 MW s energy release from metal-H_2O reaction
SPERT 1 [31] oxide core	2.6	455	17 400	155	1 800	2 200	70	Two fuel rods ruptured. Discoloration and/or deformation of 25% of fuel rods. Negligible fission-product release
	3.3	645	35 000	155	1 800	2 200	130	Two fuel rods ruptured. Discoloration and/or deformation of 25% of fuel rods. Negligible fission-product release.
SNAPTRAN 3 [32]	3.5	1 400	$\sim$ 20 000	50	$>$ 2 500	7 100	$\sim$ 4 000	Burst pressure vessel. All fuel rods ruptured, $\sim$ ½ of fuel reduced to powder form. Negligible fission-product escape.

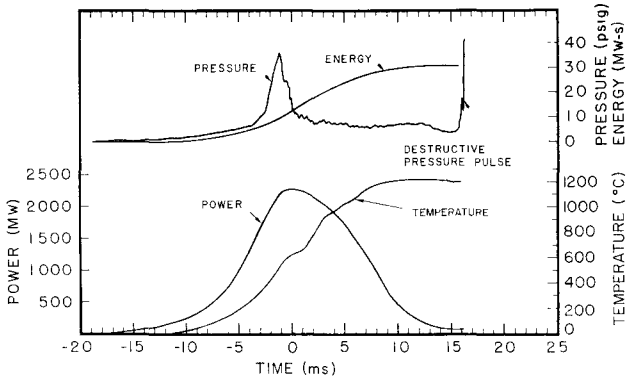


Figure 6. SPERT 1 destructive experiment—D-core

by the normal inherent shut-down mechanisms. Little recorded information is available about the nature of the explosion which destroyed the core after the fuel melted. Of the postulates which have been advanced, most of the available information supports the hypothesis of a steam explosion resulting from momentary superheating of the water in the moderator channels.

In considering the destructive effects that might result from the rupture of UO_2 fuel rods, the possibility existed that spilling or expulsion of the hot oxide into intimate contact with water could result in large pressures being generated in the core through a steam explosion mechanism. In the first destructive-region test of the oxide core, during which two fuel rods were ruptured, the expulsion of the oxide and subsequent heat transfer to the moderator acted as a threshold shut-down mechanism, as can be seen in Figure 7. Here the shut-down coefficient decreases as would be expected from a Doppler shut-down mechanism until about the time of peak power, at

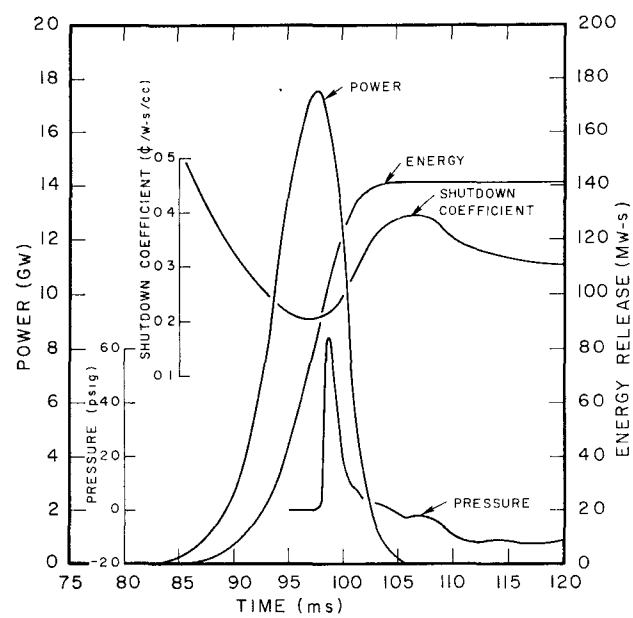


Figure 7. SPERT 1 destructive experiment — oxide core 1

which time it increases rapidly, and is accompanied by a sharp pressure increase. In this case the rapid heat transfer to the moderator acted as a prompt void-formation shut-down mechanism.

The second oxide core test, in which more reactivity was added than in the first, yielded essentially the same results. Even though the period was shorter and the peak power attained was higher, the total energy release and physical effects on the core were not greater. Figure 8 shows the power burst during the test and compares it with the predicted burst based on Doppler shut-down.

The following results were obtained from the tests: (*a*) the power excursion behavior was predictable until time of fuel rod rupture, both by calculations using Doppler reactivity feedback theory and extrapolation from previous test results; (*b*) the bursting of the fuel rods caused steam formation that aided in shut-down of the reactor and decreased the expected total energy release of the excursion; (*c*) less than 1% of the heat energy in the fuel of the ruptured fuel rods was converted into mechanical energy in the form of pressure generation; and, (*d*) the failure of a fuel rod and consequent dispersal of powdered fuel into the water during a severe power excursion did not result in pressures sufficiently large to initiate failure of additional fuel rods or seriously damage other reactor components.

The direct applicability of these results to operating oxide-fuelled reactors should take into consideration the following: (*a*) the tests were initiated from very low power levels; (*b*) there was no significant fission-product inventory, and thus no build-up of entrapped fission gases; and (*c*) the fuel used was originally intended for critical assembly work, and must be considered to be substandard compared to production fuel.

The SNAPTRAN 3 test, which resulted in total destruction of the reactor, illustrates the predictability of some threshold mechanisms; in this case, the release of hydrogen from the ZrH-U lattice when the dissociation temperature was reached. The results of the experiment were in good agreement with pre-analysis.

APPLICATION TO POWER REACTOR SYSTEMS

The excursion data can be used as handbook-type information to a limited extent, as was possible in the case of the NS *Savannah*, where the plant was very similar to one of the SPERT cores [33]. But the real value of the work is much broader and can be best indicated by outlining an approach to some of the safety considerations that enter into the design of power reactor systems. The first step in the approach is to establish the general excursion characteristics of the system. Then, limits on permissible reactivity

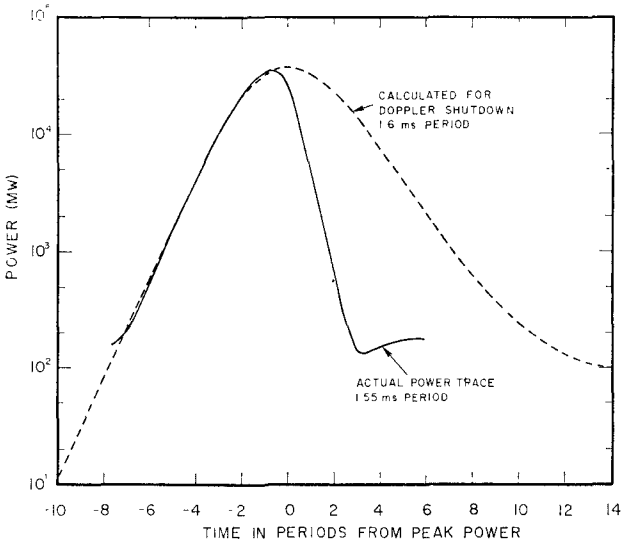


Figure 8. SPERT 1 destructive experiment — oxide core 2

increments in control or fuel units can be set. Finally, design details of mechanical and procedural controls may be determined. In this process, the economic objectives, design problems, safety requirements, and safety data interact.

Since the pressurized-water and boiling-water reactors in the USA account for a major portion of the total power reactor population, the discussion will be limited to these types. The present systems are mainly light-water cooled and moderated and use UO_2 fuel rods with either Zircaloy or stainless steel cladding. The trend toward larger cores and higher power ratings is continuing. In these reactors, three core characteristics affect the reactivity accident safety considerations in a major way: the large initial excess reactivity ($20–$30) for accommodating high burn-up and relatively large negative power coefficients; the large physical size, with attendant small neutron leakage; and, most important from the safety standpoint, the ^{238}U Doppler effect.

The first characteristic requires consideration of both the reactivity increment that can be invested in control devices or fuel bundles, and the rate at which this investment can be added to the system by moving these elements either singly or in groups. The actual numbers will vary considerably with the details of a particular design and with the detailed procedures for loading and control rod withdrawal. The fuel bundles and control blades which are used in these lattices have insertion rates in the range of several dollars per second for abnormal, accident situations involving mechanical failure of normal design safeguards that provide protection against such accidents.

These large cores, which actually contain several critical masses, have relatively weak spatial coupling of the neutron flux. Thus, two widely-spaced regions of the core may be in a supercritical or subcritical state, both possessing nearly asymptotic periods, as was demonstrated in special experiments performed in the Dresden reactor, where it took several minutes for two such regions to reach a common asymptotic period [34]. It is also theoretically possible to achieve regions of greater buckling by withdrawing control rods preferentially. The highly peaked neutron importance function results in larger incremental reactivity worths of fuel bundles and/or control elements than is the case for the large core geometry. There exists, then, the possibility of localized excursions in addition to gross core excursions involving the entire core acting as a unit.

The negative reactivity contribution of the ^{238}U Doppler effect to the self-limitation or quenching of nuclear excursions depends on the amount of resonance capture in the lattice, the oxide-fuel time constant, and the reactivity insertion rate for the excursion. Generally, for relatively short-period excursions the Doppler coefficient is the major shut-down mechanism for these reactor types, since the low values of the thermal diffusivity for the oxide fuel limits the rate at which heat can be transferred out of the fuel and, in turn, results in increased fuel temperature and a corresponding negative reactivity effect. The oxide core tests provide reference points for evaluating the characteristics of the Doppler shut-down mechanism.

In the dynamic response of these cores to nuclear excursions, such parameters as neutron lifetime, delay fraction, reactivity coefficients, scram reactivity insertion rates, etc., are important. The values of these parameters vary considerably with the details of the design. However, from the available published information (for typical lattices) [35–38], the ranges of these parameters are the following: the ratio of the delayed neutron fraction to the prompt neutron lifetime, which is defined as α_R, 100 to 750 s^{-1}; the power coefficient, -2 to $-5c/\%$ change in power; the moderator temperature coefficient, 0 to $-5 \times 10^{-4}\Delta k/°F$; the moderator void coefficient, 0 to $-3 \times 10^{-3}\Delta k/\%$ void; the Doppler coefficient, -1 to -5×10^{-5} $\Delta k/°F$; the scram reactivity insertion rate, $10 to $20/s average rate including delay time, with peak rates two or three times greater.

On the basis of these characteristics and the present understanding of Doppler coefficients and excursion kinetics, the principal features of nuclear excursions for these cores can be estimated, as shown in Table 3. Note that it takes an energy density of approximately 1.1×10^4 watt seconds/cm^3 to melt UO_2, and approximately two to three times that to vaporize UO_2, assuming that the UO_2 is initially at room temperature. In the low-ramp-rate range there is no fuel melting, the periods are relatively long, and significant heat transfer to the moderator occurs during the burst. The shut-down mechanisms are the Doppler effect and the negative moderator coefficient. The expected

Table 3. Excursion parameters for pressurized-water and boiling-water reactors

Range	Reactivity insertion rate ($/s)	Equivalent step ($)	Minimum period (ms)	Energy density (av) (W s/cm² × 10⁻⁴)	Energy density (peak) (W s/cm² × 10⁻⁴)	Principal shut-down mechanismus
Low	< 2.5	< 1.8	> 4	< 0.14	< 0.44	Doppler effect, negative moderator coefficient
Medium	2.5–25	1.8–3.5	4–1.2	0.14–0.48	0.44–2.2	Doppler effect
High	> 25	> 3.5	< 1.2	> 0.48	> 2.2	Doppler effect, core disassembly

consequences of such an excursion are generally trivial, although in some cases the scram system must be relied upon to prevent clad damage [35, 36]. The medium range has faster ramp rates, shorter periods, and only a small amount of heat transfer to the moderator during the burst. Shut-down is predominantly from the Doppler effect. At the upper end of this range, there may be a small amount of fuel clad failure during the burst with fuel (UO_2) melting and, possibly, vaporization of UO_2. After the termination of the burst, further clad failures may occur at a relatively slow rate while the energy stored in the fuel is transferred to the clad and the water. No significant mechanical effects or transient pressures are postulated for excursions in this range. In the high range (very fast ramp rates and very short periods), the predominant shut-down mechanism is the Doppler effect, possibly supplemented by fuel dispersion and moderator expulsion during the burst. Significant fuel melting and vaporization could occur with, possibly, substantial mechanical effects and transient pressures, if the stored energy in the fuel can be transferred rapidly to the water. No experimental data are available on these high-ramp-rate accidents. However, the most extreme tests shown in Table 2 extend well into the medium-ramp-rate range. The results of these tests generally confirm the qualitative differences between the low-ramp-rate and medium-ramp-rate ranges discussed above.

In the above discussion, the ranges for the energy release and consequences were established on the basis of experimental excursion studies, analysis of kinetic behavior, and the lattice characteristics. However, the credibility, or even possibility, of achieving the postulated excursions was not considered. The physical limits on attainable reactivity insertion rates [15], and the inherent safety characteristics of the particular system, must be considered in relation to the following principal types of accidents: (*a*) mechanical malfunction of the control system, causing rapid removal of the control rods; (*b*) inadvertent, rapid, manual removal of control rods; (*c*) procedural operator error in withdrawing control rods, or in changing control-rod patterns during operation (e.g., the classical start-up accident); (*d*) fuel-handling or fuel-loading accident, including potential accidents associated with handling of temporary poisons in the form of sheets or strips [37]; (*e*) accidents arising from moderator effects, such as cold-water accidents in PWRs and BWRs, and positive-pressure ramps in BWRs; and, (*f*) accidents associated with malfunctions of the soluble-poison control system, resulting in accidental removal of the poison or insufficient shut-down margin in the liquid poison system and leading to a refuelling accident.

Both the potentiality for occurrence of such accidents and the possible magnitude of the consequences can be reduced by limiting the reactivity increment and insertion rate in fuel or control units. In the design of this type of power reactor these limits are significantly affected by safety considerations. Important factors which enter into the specification of the mechanical control system are reactor shut-down margin, maximum control-rod worth, and scram requirements. Once design criteria have been established for the first two these factors (which tend to be limiting), the excess reactivity which may be loaded into the reactor core for a given control-rod spacing and the cold-to-hot operating reactivity swing for the fuel may be determined. The key to interrelating the fuel cycle requirements and the movable control system is an approximate relationship between thermal control-rod strength, thermal diffusion length, and control blade (span/pitch) ratio [39], which is essentially a restatement of absorption area theory [40].

With proper safeguards design and operating procedural limitations for the plant, most of the potential excursion accidents in the above categories fall into the low range (no fuel damage). For example, the classical start-up accident leads to reactivity insertion rates of the order of cents per second to ten cents per second [35–38]. Accidents that are outside this range can be conceived by postulating compound violations of procedural controls with concurrent, unrelated mechanical or equipment failures.

SUMMARY

The excursion studies have shown that for many types of reactor systems excursion behavior is well understood and that the self-limiting capabilities can be sufficient to provide protection against substantial reactivity disturbances. The studies have been principally devoted to systems with relatively strong quenching properties and under initial test conditions such that substantial energy releases would be required to cause damage. Thus, neither the effects of locally positive reactivity coefficients nor excursions initiated under high power conditions have been examined. The data from excursions that resulted in mechanical damage have provided information on the shut-down mechanisms that operate under such conditions, the magnitude of the transient pressure effects, and the radiological consequences.

In considering the possible sources of the maximum credible accident, attention must be given to the means of initiation and credibility of occurrence of reactivity accidents, including simple plant failures. However, the body of excursion data and the present state of understanding of reactor excursions are such that the reactivity accident, for many power reactor systems, can, through appropriate design, be reduced to secondary importance in the consideration of the maximum credible accident. Of particular importance to many present day designs of pressurized-water and boiling-water reactors is the fact the light-water oxide-core power burst data can be reproduced analytically using models developed from first principles. This lends confidence to the model predictions of the excursion behavior of power reactor cores which are reasonably similar to the experimental configurations. Information on the consequences of very short period (high reactivity range) excursions in these cores is limited since this problem is not very amenable to analysis and only a small amount of experimental data is available. In particular, no data are available to determine if there exists a destructive threshold for these cores similar to that observed in other systems, although present information does not indicate that such a threshold occurs with the rupture of a few fuel rods. Information in this general area can be useful to the reactor designer in optimizing nuclear system design from the standpoint of safety and performance.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/283 Etats-Unis d'Amérique

Etude des sautes de puissance d'un réacteur par W. E. Nyer *et al.*

Il n'est pas exclu que les sautes de puissance dans les réacteurs nucléaires puissent, dans des cas extrêmes, causer des dommages dans l'installation, et libérer des quantités importantes de produits de fission; leur étude est donc un aspect important de l'évaluation de la sûreté d'un réacteur de puissance. Lorsque l'on fait cette évaluation, il faut déterminer les mécanismes physiques intrinsèques qui ont pour effet d'arrêter ou d'amplifier la saute de puissance, déterminer aussi leur comportement pendant la saute de puissance et bien comprendre leur effet sur l'allure de la saute de puissance. Ces mécanismes intrinsèques, qui sont en partie des propriétés du type de réacteur, peuvent être sensiblement modifiés par des caractéristiques particulières. Il faut examiner soigneusement ces caractéristiques pour s'assurer que certains détails du projet n'ont pas d'effets indésirables, pour évaluer les moyens possibles de provoquer des sautes de puissance accidentelles, et enfin pour fixer certaines spécifications et mesures de contrôle qui devront s'appliquer au fonctionnement de l'installation aussi bien qu'à la conception elle-même.

On dispose maintenant de nombreuses indications touchant la sûreté des réacteurs de puissance du point de vue des sautes de puissance grâce aux études expérimentales faites dans des conditions diverses avec des réacteurs de différentes filières: TRIGA, qui est ralenti au ZrH; TREAT, qui est ralenti au graphite; KEWB, qui est homogène et aqueux; Godiva, qui est un assemblage entièrement métallique à neutrons rapides; SPERT 1, qui est ralenti à l'eau et dont le combustible est en aiguilles ou en plaques; SPERT 2, qui est ralenti à l'eau lourde et dont le combustible est en plaques; SPERT 3, qui est ralenti à l'eau à haute pression et dont le combustible est en plaques. On a pu démontrer que les divers mécanismes d'arrêt étaient efficaces dans le cas de l'autolimitation des sautes de puissance provoquées délibérément, et mesurer leur effet quantitatif sur la réactivité. Ces mécanismes étaient la production de gaz par radiolyse, des modifications (dues à la chaleur) dans la géométrie de la configuration du combustible, des variations de la densité du fluide de ralentissement et de refroidissement, des variations des sections efficaces de diffusion, les effets Doppler et dans plusieurs cas, le désassemblage du réacteur. La plupart de ces mécanismes sont importants dans les types courants de réacteurs de puissance; toutefois, l'effet quantitatif sur la réactivité varie selon le type de réacteur et même selon les conditions de fonctionnement de l'installation (comme l'ont montré les expériences faites avec SPERT 2 et SPERT 3). Comme on s'y attendait pour ces réacteurs, une augmentation de la température et de la pression initiales au cours d'une série d'essais a eu pour effet un accroissement significatif, mais non extrême, de l'énergie libérée. Des débits élevés du fluide de refroidissement ont également eu pour effet un accroissement de l'énergie libérée, mais non de la température du combustible. Qui plus est: lorsque l'on a simulé les conditions qui règnent dans les réacteurs de puissance, on n'a pas fait décroître sensiblement la capacité qu'ont ces installations de limiter elles-mêmes les sautes de puissance provoquées par des additions de réactivité considérablement supérieures à celle que provoquent les neutrons instantanés. Les résultats obtenus avec tous les systèmes considérés indiquent une remarquable similitude dans la façon dont la forme de la saute de puissance, la puissance de crête et l'énergie varient en fonction de la grandeur de la réactivité ajoutée, et les descriptions mathématiques en sont remarquablement simples. On ne pense pas que ces caractéristiques soient qualitativement différentes dans les réacteurs de puissance. Ce qui est particulièrement important, c'est que dans les expé-

riences TRIGA, TREAT et SPERT 1 avec cœur à oxyde d'uranium, on peut calculer directement la grandeur de l'effet d'arrêt et les détails de l'allure de la saute de puissance à partir de considérations fondamentales.

L'accroissement en énergie fournie par la saute de puissance qui s'est produit au cours des essais avec le cœur à oxyde d'uranium dans le réacteur SPERT 1, où le système de fixation du combustible permettait aux barres de combustible de se courber, est un exemple de l'influence que peuvent avoir sur la sûreté des réacteurs certains détails de conceptions et indique la possibilité de tirer profit de cet effet.

On a observé des différences et des similitudes importantes dans les résultats de séries d'essais destructifs exécutés dans le réacteur SPERT 1 avec du combustible en plaques et en aiguilles. La fusion de 3% des plaques de combustible au cours d'une saute de puissance n'a provoqué qu'une faible pression transitoire; mais la fusion de 35% au cours de l'essai destructif a provoqué des pressions de l'ordre de plusieurs milliers de livres par pouce carré, 10% environ du total de l'énergie libérée pouvant être attribués à des sources chimiques, et une proportion inférieure à 3% apparaissant sous une forme mécanique. Les effets puissants de désassemblage mécanique n'ont pas contribué à l'arrêt de la saute de puissance. En revanche, la rupture des aiguilles de combustible au cours de l'essai avec un cœur à oxyde faiblement enrichi a eu un fort effet d'arrêt, qui a modifié la forme de la saute de puissance et limité la libération d'énergie. Malgré l'addition d'une plus grande réactivité et une plus grande libération d'énergie, les effets mécaniques ont été pratiquement négligeables.

A/283 США

Отклонения в режиме работы реактора

У. Э. Найер

Вследствие того что отклонения в режиме работы ядерных реакторов в крайних случаях могут повлечь за собой повреждение установки и освобождение значительных количеств продуктов деления, изучение отклонений в режиме работы является важным фактором в оценке общей безопасности системы энергетического реактора. При этом должны быть определены свойственные реактору физические особенности, которые замедляют отклонения в режиме или способствуют ему. Эти факторы, отчасти определяемые свойствами данной реактивной системы, в то же время сильно зависят от особенностей конструкции. Поэтому следует убедиться, что нежелательные эффекты не обусловлены конструктивными деталями, оценить вероятность появления случайных отклонений в режиме работы и, наконец, выработать определенные требования к конструкции установки и методам ее эксплуатации.

Значительная информация о безопасности энергетических установок при отклонениях в режиме работы получена в результате опытов, проведенных в различных условиях с различными реакторными системами, такими, как TRIGA (реактор фирмы «Дженерал атомик») — система с замедлителем из гидрида циркония; TREAT — система с графитовым замедлителем; KEWB — водная гомогенная система; реактор «Леди Годива» — цельнометаллическая сборка на быстрых нейтронах; SPERT-I — система со стержневыми или пластинчатыми тепловыделяющими элементами и водяным замедлителем; SPERT-II — система с тяжеловодным замедлителем и пластинчатыми тепловыделяющими элементами и SPERT-III — система высокого давления с пластинчатыми тепловыделяющими элементами и водяным замедлителем. Было показано, что различные механизмы замедления отклонения эффективно органичивают искусственно вызванные отклонения мощности. Определено их влияние на реактивность. Этими механизмами были: радиолитическое образование газа, геометрические изменения в конфигурации тепловыделяющих элементов из-за нагрева, изменения плотности теплоносителя-замедлителя, в результате которого произошли изменения механизмов теплопередачи, изменение сечений рассеяния, доплеровских эффектов и в некоторых случаях разборка реактора. Большинство этих механизмов играет важную роль в работе современных типов энергетических реакторов, но величина их воздействия на реактивность зависит от конкретного типа реактора и даже от режима работы установки (это было показано в опытах с реакторами SPERT-II и SPERT-III) Как и следовало ожидать, увеличению начальных температур и давлений в ряде испытаний этих систем соответствовало значительное, но не чрезмерное увеличение выделения энергии Интенсивные потоки теплоносителя также вызвали рост выделения энергии, но не увеличение температуры топлива. Однако наиболее важным является то, что моделирование условий энергетического реактора не уменьшило значительно способность этих установок к безопасному самоограничению отклонений мощности, вызванных введением реактивности, значительно выше мгновенно критической. Результаты, полученные для всех систем, обнаружили хорошее сходство в зависимости формы вспышки, максимальной мощности и энергии от введенной реактивности, которая очень просто выражается математически. Эти особенности не будут качественно различными в случае энергетических реакторов. Особенно важно, что

в опытах с реакторами TRIGA, TREAT и SPERT с активными зонами из окиси урана величина эффекта замедления и детали хода вспышки могут быть вычислены непосредственно из основных соотношений.

Повышение выхода энергии при отклонении, имевшее место в опытах на реакторе SPERT-I с активной зоной из окиси урана, имеющей систему опор топливных стержней, которая позволяла осуществлять их изгиб, является примером влияния проектирования деталей на безопасность реактора, благодаря которому имеется возможность полезного использования данного эффекта.

Важные различия и сходство результатов наблюдались при проведении серии испытаний на разрушение в реакторе SPERT-I активных зон с пластинчатыми и стержневыми топливными стержнями. Расплавление 3% топливных пластин во время отклонения мощности привело к незначительному повышению давления, в то время как расплавление до 35% топливных пластин в опыте по разрушению привело к образованию давления, приблизительно равного 100 *ат*, причем около 10% выделенной энергии обусловлено источниками химической и не более 3% — механической энергии. Значительный механический демонтаж активной зоны не способствовал ослаблению вспышки энергии. Напротив, разрыв топливных стержней в опыте со слабообогащенными активными зонами из окиси урана способствовал появлению сильного замедления, которое изменило форму вспышки и ограничило выход энергии. Несмотря на большую введенную реактивность и больший выход энергии, механические эффекты были незначительны.

A/283 Estados Unidos de América

El comportamiento de un reactor en una excursión de potencia

por W. E. Nyer *et al.*

La consideración del comportamiento de un reactor en una excursión de potencia es un aspecto importante de la estimación de la seguridad general de un reactor de potencia, ya que es concebible que de tales excursiones se deriven, en casos extremos, daños a la central y el escape de cantidades importantes de productos de fisión. Al hacer esta evaluación, hay que identificar los mecanismos físicos inherentes al reactor, que actúan en contra o a favor de la excursión, fundamentar su comportamiento durante la excursión y comprender su efecto sobre la marcha de la misma. La influencia de las características del proyecto sobre estos factores, que en parte son propiedades del tipo de reactor, es muy grande. Aquéllas deben examinarse con cuidado para impedir que los detalles del proyecto introduzcan efectos no deseables, para estimar los mecanismos posibles por los que se iniciarían excursiones accidentales, y finalmente para establecer ciertas especificaciones o normas de control a aplicar en la explotación de la central o en el propio proyecto.

Se dispone en la actualidad de gran cantidad de información sobre las excursiones de los reactores de potencia, deducida de los ecperimentos llevados a cabo en muchas condiciones distintas con gran número de reactores diferentes, a saber: TRIGA, sistema moderado por ZrH; TREAT, moderado por grafito; KEWB, sistema homogéneo acuoso; Godiva, conjunto rápido completamente metálico; SPERT 1, moderado por agua y con placas o barritas como combustible; SPERT 2, moderado por agua pesada y con combustible laminar; y el SPERT 3, que es un sistema de agua a alta presión con combustible laminar. Se ha probado la efectividad de diversos mecanismos autolimitadores en excursiones de potencia intencionadas; también se midió la magnitud de sus efectos sobre la reactividad. Estos mecanismos fueron: la formación de gases por radiólisis, los cambios geométricos en la configuración del elemento combustible a causa del calentamiento, las variaciones de las densidades del combustible y del moderador a causa de los diversos mecanismos de transmisión del calor, las variaciones de las secciones eficaces de dispersión, los efectos Doppler y, en varios casos, el desmantelamiento del reactor. La mayoría de estos mecanismos son importantes en los tipos actuales de reactores de potencia; sin embargo, la magnitud del efecto de reactividad depende del tipo de reactor e incluso de las condiciones de explotación de la central (como se demostró con experimentos realizados en los reactores SPERT 2 y SPERT 3). Como era de esperar en estos sistemas, al aumentar la presión y la temperatura iniciales en una serie de pruebas, aumentaba en cantidades importantes, pero no extremas, la energía liberada. Al aumentar el caudal de refrigeración aumentaba también la energía liberada, pero no aumentó la temperatura del combustible. Tiene mucha importancia el que al simular las condiciones de un reactor de potencia no disminuya mucho la capacidad de estas centrales para autolimitar, con seguridad, las excursiones producidas al insertar cantidades de reactividad muy superiores a las necesarias para que el reactor alcance la criticidad sin los neutrones retardados. Los resultados de todos los sistemas son notoriamente similares en el modo de variar, con la reactividad introducida, la forma de la curva de potencia, el valor máximo de la misma y la energía liberada; además, las funciones correspondientes se pueden expresar mediante fórmulas matemáticas sencillas. No se espera que estas características sean cualitativamente distintas en los reactores de potencia. Es particularmente interesante hacer notar que en los experimentos

con núcleos de óxido llevados a cabo en el TRIGA, el TREAT y en el SPERT 1, se supo calcular directamente, a partir de consideraciones fundamentales, la magnitud del efecto autolimitador y los detalles de un aumento brusco de la potencia.

El aumento de la energía liberada en la excursión que tuvo lugar en el SPERT 1 durante las pruebas con núcleo de óxido con un sistema de soporte del combustible que permitía el pandeo de las barras del mismo, es un ejemplo de la influencia que los detalles del proyecto pueden tener en la seguridad del reactor y sugiere la posibilidad de utilizar el efecto con ventaja.

Se observaron diferencias y semejanzas importantes en los resultados de una serie de pruebas destructivas en el SPERT 1 con núcleos con placas y barritas combustibles. La fusión de un 3% de las placas combustibles en una excursión dió lugar a un transitorio poco importante de la presión, mientras que la fusión del 35% en una prueba destructiva dió lugar a presiones de centenares de atmósferas, siendo de origen químico el 10% de la energía total liberada y apareciendo en forma mecánica no más del 3%. Los efectos del gran desmantelamiento mecánico producido no contribuyeron a la autolimitación del estado transitorio de la potencia. Por el contrario, la ruptura de las barritas de combustible en la prueba del núcleo de óxido poco enriquecido dió lugar a un fuerte efecto de autolimitación que cambió la forma del transitorio y limitó la cantidad de energía liberada. A pesar de que la inserción de reactividad y la energía liberada eran mayores, los efectos mecánicos fueron prácticamente despreciables.

Fast-reactor safety research

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In 1958, it was stated [1] that investigations to consider the fuel redistribution and reassembly problems of fast-reactor melt-down formed the focus of fast-reactor safety studies. This is still the case in 1964. Many questions have been answered in the intervening years, particularly with respect to the likelihood and possible results of a melt-down in the current generation of metal-fueled fast reactors, viz., the EBR-II and the Enrico Fermi reactors. However, the emphasis in fast-reactor design studies has shifted to large cores, non-metal fuels such as oxides, carbides, and cermets, and the use of plutonium and ^{233}U instead of ^{235}U. This shift in emphasis has necessitated studies of new melt-down-initiating phenomena peculiar to the core designs involved, together with theoretical and experimental attacks on questions relating to the possibility of reassembly and to events during the power burst resulting from reassembly.

MELT-DOWN-INITIATING PHENOMENA

Studies of fast-reactor melt-down begin with the assumption that conditions somehow arising in the core lead to rapid melting of a large part of the fuel.

Mechanical mal-operation

The worst melt-down accidents studied have been the result of hypothesized mechanical mal-operations, such as unchecked control rod withdrawal, fuel-handling device failures, and failure of all pumps coincident with failure of all safety rods. To a certain extent the possibility of such mal-operation exists in all reactor types; the possibility has been of particular concern in fast-reactor design because of the relatively small amount of experience available on sodium-cooled systems. As experience accumulates it is becoming apparent that the reliability of mechanisms used in sodium can eventually be just as good as that of those used in other coolants [2–4]. Successful high-power operation of the EBR-II and Fermi reactor will go far toward moderating the present tendency to assume the worst about the mechanical behavior of fast-reactor components.

Fuel sub-assembly melting

It has been considered impossible to preclude completely the melting of a single fuel sub-assembly in a core composed of a large number of such sub-assemblies. This concern is to a certain extent due to the lack of experience regarding the behavior of fuel under actual fast-reactor conditions. The assumptions regarding safe exposure level before excessive swelling occurs are based on tests in thermal reactors, although some experience on a fuel like that of the Fermi reactor has been gained in the Dounreay reactor [4] and on an oxide fuel in the Russian BR-5.

The melting of a single sub-assembly might lead to a gross melt-down due to two possible chains of events. The melted fuel from the first sub-assembly could cause plugging in adjacent sub-assemblies, resulting in progressive melting. The second, and much more likely chain of events, is that the motion of melted fuel could cause an increase in the reactivity of the core as a whole. Whereas the worst reactivity increase of the ^{235}U fueled, 105 sub-assembly Fermi reactor due to this phenomenon is given as about 0.5 dollar [5], it should be noted that a plutonium core in the same reactor would cause a reactivity increase of more than 1.5 dollars. In a reactor with a larger number of fuel sub-assemblies than Fermi, the reactivity available from rearrangement of the fuel in a single sub-assembly would be proportionately less.

Internal reactivity increasing effects

Calculations have indicated that the reactivity effect of a sodium density reduction may be positive over the whole core or in part of many cores of current interest [6]. This observation has added to the existing concern over the unstable behavior of sodium boiling [1] since boiling in such a reactor would cause a reactivity increase before or during the melting of fuel in a sub-assembly. One design study [7] utilized a pressurized primary system to increase the boiling temperature

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above the sodium temperature that would be reached during transients.

In 1958, interest in the Doppler effect was principally directed toward showing that the positive effect in ^{235}U was not dangerously large. Theoretical [8], and experimental [9, 10], work has since dispelled these fears. The Doppler effect is still of great importance in considerations of fast-reactor safety [11], both for highly enriched cermet fuels, in which the effect is presumably positive, and for large reactors diluted with ^{238}U in which the effect is negative. In the latter case the negative Doppler coefficient can be of great value in preventing melt-down [12–14]. Recent theoretical and experimental work on the Doppler effect in fast-reactor designs of current interest has been reported elsewhere in these proceedings [15–17].

Sudden and large reactivity increases in oxide cores may result if the oxide gradually expands due to burn-up and then melts and collapses suddenly within the cladding due to a temperature or power transient. Determinations of the likelihood of such an effect and of the value of various preventive measures is one of the objectives of the FARET and SEFOR programs [18, 19].

Reactor instability

A sustained effort has been applied to the study of the oscillatory and excursion behavior of fast reactors. In 1958, the existence of oscillatory instability on EBR-I Mark II had been reported [20] and a possible feedback model which would cause the instability had been suggested by Kinchin [21] and elaborated on by Bethe [22].

A more complete feedback model, relating the thermal properties of the reactor to the feedback time constants, has been formulated by Storrer [23]. Further elaboration of this model by R. R. Smith *et al.* [24], has led to an understanding of the mechanical features of EBR-I that caused the observed behavior. Substantiation of this is presented by the stable operation of later EBR-I cores, which were designed using the knowledge obtained from the analysis of the Mark II core. Several authors have considered the stability and kinetics of fast reactors. H. A. Sandmeier performed an analysis in which he considered the effects of nonlinearities in the kinetics equations and in the feedback behavior on reactor kinetics and stability [25]. Carter [26, 27] also has considered nonlinear effects in kinetics and feedbacks. The feedback relations in fast reactors have been analysed by Storrer [28] and he has developed a method for determining power coefficients and associated time constants by use of an asymptotic technique. Recently Wakabayashi reported an analytical study of the EBR-I Mark IV transfer function that closely predicts experimental behavior [29].

To establish general principles for the design of stable reactors, several studies have been made. In one series of studies [30–35], the simultaneous nonlinear differential equations describing the power in a reactor having a prompt positive reactivity coefficient and a delayed negative feedback due to heat conduction, were integrated numerically for a limited range of reactor parameters. The calculations show that, while such a reactor may be unstable above some power level, a step insertion of reactivity made at a sufficiently low power level will result in a power excursion of limited extent. In another study [36], a similar feedback model was used and the behavior of a reactor with large time constants typical of ceramic fuel was compared with that of a reactor with shorter time constants typical of metal fuel. These studies indicated that the oscillatory behavior of a reactor with a positive power coefficient component due to a gross negative material worth of sodium should not present serious problems as long as the reactor has reasonably large negative feedback components, even if the negative components have time constants larger than that of sodium.

In many instances, a reactor may be stable in the normal operating range when oscillated, but prompt positive feedback effects may be harmful during certain excursions. Several reactor designs have been evaluated as to this characteristic and the results have been reported [12, 13, 37].

The excursion behavior of a reactor with a positive sodium coefficient and a negative Doppler coefficient has been investigated for loss of coolant, loss of flow, and ramp reactivity insertion [38]. For the configuration studied, all three types of excursions could destroy the reactor. However, when the reactor design was modified to include 8.3 vol % BeO, the Doppler coefficient was doubled and the sign of the sodium coefficient for the reactor as a whole was changed, although expulsion of sodium from the center of the core still resulted in a positive reactivity effect. The modified reactor would survive the excursions considered.

To overcome safety problems during credible excursions, some of the groups that participated in the 1 000 MW(e) Fast Reactor Design Study [7, 39, 40] selected designs which would respond favorably during excursions by using design innovations.

It seems reasonable to expect that as fast reactor cores get larger, difficulties may arise with local instability due to loose coupling of different core regions. No space and time dependent kinetics evaluation of this problem has been reported.

INVESTIGATION OF CONSEQUENCES OF GROSS FUEL MELTING

Gross fuel melting cannot at present be excluded from consideration in safety studies for fast reactors of arbitrary core type and size. For this reason experimental and theoretical work has been conducted to investigate the consequences of such melting.

Melt-down experiments

Analysis of a melt-down requires knowledge of the conditions producing failure, the modes of failure and core material movement, rates of movement, degree of coherence in the movements, and reactivity effects caused by the movements. Many phenomena are involved, including fuel slumping, dissolution of cladding by molten fuel, expulsion of fuel from cladding, effects of fission gases, physical interactions between pins in clusters, coolant boiling, and expulsion. For reactor designs with appreciable positive sodium-void reactivity coefficients, such as those for some large plutonium-fueled cores [38, 41], the rates of coolant expulsion must be considered for their effects on reactivity as well as on fuel movement. Coolant expulsion phenomena may be particularly complex for some larger core designs, in which local sodium void coefficients may vary widely in magnitude from one point to another, and may even change sign.

Out-of-pile experiments

Melting has been studied by performing out-of-pile tests on irradiated and unirradiated pins, using electrical heating [42–45]. The pins used in these studies were uranium alloy pins of the type used in the Enrico Fermi reactor and simulators of these pins. Electrically heated unirradiated uranium-molybdenum alloy fuel pins clad with zirconium were tested in an argon atmosphere to determine clad containment capability for molten fuel [42, 43]. Various clad thicknesses were considered. In tests in which the fuel was 90 °F above its melting point, the clad would contain the fuel material from 30 to 48 min depending upon the thickness and original condition of the clad. Upon failure of the clad, the uranium-molybdenum alloy flowed along the surface of the clad at about 5 in/s. The only exception occurred when a second hole formed above the first failure, allowing argon gas to enter to balance the gas pressure so that fuel hydrostatic pressure would cause the molten alloy to jet out of the bottom hole in the clad.

Irradiated fuel pins have been tested [44], using resistance heating to investigate the mode of failure of pins exposed to increasing temperatures. The pins that were used had been irradiated to burn-up from 0.2 to 0.9 total per cent burn-up at temperatures from 700 to 1 350 °F. The pins were subjected to temperature excursions of 125, 1 800, and 5 400 °F/s until pin failure occurred. Irradiated fuel pins failed by a sequence of swelling, cladding rupture, and the expansion of molten fuel alloy out of the cladding when heated above the melting point. The molten fuel alloy was very porous and had a low density. For burn-ups from about 0.2 to 1.0 at.% at about 2 300 °F, the volume of the molten metal was as much as three times the original volume. The molten fuel had little fluidity due to its spongy nature.

These results are significant from the standpoint of limiting the maximum energy release potential of the core due to melt-down. They indicate that a relatively small inventory of fission products could lead, upon melting, to a fuel density decrease of a factor of as much as three. Accordingly, the coolant volume, which about equals the volume of the unswollen fuel, would be filled and subsequent fuel expansion would decrease the density of the core. The spongy nature of the molten material thus would tend to retard or even prevent the assembly of a high density mass of core material. Moreover, even if it is assumed that a critical mass could be assembled, the low fluidity of the material would decrease the rate of assembly and hence the rate of reactivity increase.

To assess the role played by the coolant in causing a melt-down and in relocating the fuel material following a melt-down, an out-of-pile simulation test was carried out in which water and lead were used as simulants for sodium and uranium alloy, respectively [45]. A four-pin unit cell, heated electrically, was placed in a glass-walled flow loop, and the conditions of the coolant and simulated fuel during and after various flow-failure incidents was observed. In none of the tests did the fuel mass compact in a more potentially dangerous manner than that assumed in a Bethe-Tait analysis [46]. In general, when rates of fuel motions exceeding the rate that would result from gravity alone were observed, the motion was in the direction of spreading rather than compacting the fuel mass. In most tests, the initial migration of simulated fuel material was away from the core midplane, reducing reactivity. According to the tests, subsequent coolant flow may reintroduce part of this reactivity, but not at a rate faster than the model of gross slumping under gravity. The reactivity of the final disposition of fuel in most tests appeared to be less than that of the original configuration.

In no test did slug flow occur; flow was more like that of a slurry in which individual fuel globules are visible. This indicates that the slug flow usually assumed in a melt-down energy release analysis may lead to unrealistically pessimistic results.

Although these studies did not take into account all the forces and processes which are thought to exist during a reactor melt-down, they have furnished evidence that the assumptions generally used in the analysis of the energy release during melt-down are reasonable. More elaborate experiments are necessary in order to determine the behavior of fuel material during a melt-down.

In-pile experiments

The in-pile experiments have consisted of a continuing program in the Transient Reactor Test Facility

(TREAT) to study melt-down phenomena and their interactions [47]. The basic philosophy is to separate the various problems, such as cladding penetration, rates of heating and fission gas effects and to study them as systematically as possible. A key phase of the program consists of a theoretical investigation of relevant phenomena, the analysis of experimental results, and calculations relating results to reactor safety analyses.

Major types of special equipment developed for use in TREAT include simple graphite-lined steel capsules [48], a transparent melt-down facility for photographic experiments [49], modifications of opaque and transparent capsules for tests on irradiated specimens [50, 51], stagnant sodium capsules for EBR-II size samples [52] and for larger pins [53], and a sodium loop designed to fit inside a TREAT fuel element [54]. In addition, a large sodium loop is being constructed for experiments on melt-down of clusters in flowing sodium. A major tool now under development is a device using prompt neutrons for observing fuel movement during experiments in which optical photography cannot be used [55].

TREAT-related studies began with experiments on metallic uranium fuel elements [47]. Ceramic specimens were used as they became available from the various developmental programs. The major part of the TREAT program thus far has been concerned with the investigation of modes of failure and types of fuel motion for single pins exposed in an inert gas atmosphere. Conditions that have not been studied yet, or have been surveyed only briefly, include: cluster interactions, fission gases in ceramic fuels, behavior of high burn-up uranium metallic fuels, flowing sodium environment, and metallic thorium fuel. Sufficient data have been obtained to define failure characteristics for a variety of conditions, and to permit analytical studies to be made on the coherence of failure of a reactor like EBR-II [56].

Results of TREAT experiments

Failure thresholds, modes of failure, and mechanisms of fuel movement were studied for unirradiated EBR-II Mark I samples, refractory metal clad EBR-II pins, and Fermi Core A specimens.

EBR-II pin results were consistent with a simplified model incorporating calculated cladding strength and internal pressure, and cladding penetration rate data [47]. Satisfactory correlation was found between EBR-II pin failure times as measured in TREAT [57], and pin failure times obtained in out-of-pile dissolution rate experiments [58]. Results from refractory metal clad specimens were consistent with extrapolation of refractory metal penetration data [59], assuming a diffusion controlled process [60]. EBR-II Mark I sample behavior may be summarized briefly as follows: A fuel-cladding interface temperature of about 1 020 °C marks the threshold for failure by dissolution of the cladding, with typical penetration times of about one second. At an interface temperature of about 1 200 °C, cladding failure occurs by bursting caused by the internal pressure of the bonding sodium and the inert gas in the expansion space above the sodium. Following failure, fuel material is ejected by the internal pressure.

Fermi A fuel pins had a threshold, due to minor failures in the cladding at a cladding temperature of about 1 300 °C. At higher temperatures, attack of the cladding becomes apparent in agreement with extrapolations of penetration data [59]. Fuel movements are consistent with the effects of gravity alone [47, 61].

Failures of EBR-II Mark I elements exposed in stagnant sodium capsules are correlated in the same fashion as dry samples failures [62].

Experiments on dry EBR-II Mark I and Fermi A pins, previously irradiated to produce appreciable inventories of fission products, demonstrate sufficient short-term fuel swelling under melt-down conditions and melt-down products of sufficiently low density to play a significant role in affecting reactivity changes [63, 64]. The TREAT flux-time integral required for failure of irradiated Fermi pins is about 20 % less than for unirradiated specimens. This effect was not seen in EBR-II pins that had a stronger cladding.

The movement of fuel prior to failure and the phenomena of failure were studied for dry EBR-II size gas bonded UO_2 pins. Fuel expulsion under pressure of the bond gas can occur [65]. Phenomena occurring before cladding failure have also been studied for larger oxide specimens run in stagnant sodium [53, 66]. Changes in fuel behavior produced by substituting uranium sulfide for UO_2 in EBR-II size elements have also been surveyed [67]. The principal differences found were that the uranium sulfide showed less fracturing but more plastic deformation.

Use of TREAT results in melt-down analysis

In an actual reactor, the possible rates of reactivity change in a melt-down accident are determined by the degree of coherence in fuel failures, as well as by the physical phenomena producing material movement. TREAT data on EBR-II Mark I fuel were used in theoretical analyses of the times of failure in an EBR-II-like reactor [56] for the following three hypothetical accidents: pumping power failure without scram; blocked sub-assembly inlet orifice; and central fuel sub-assembly dropped into a reactor critical at low power. The calculations used the feedback characteristics of intact fuel. However, spatial variations in power and coolant flow, transient heat transfer, and propagation of failure from one pin to an adjacent one were considered. For the first case, with constant power, the spread in failure times is about three seconds. The

second accident has a spread of about 0.5–1.0 s for simultaneous blockage of the flow to the entire core. In the third accident, the spread is decreased to about 0.04 s. For a greater time spread one would expect less chance of the entire core collapsing as a unit. In the case of a spread as short as 0.04 s, the melting rate would not limit the assembly rate and the possibility of large rates of reactivity increase exists.

A study has been made of the interrelated phenomena involved in transient pressure generation and coolant expulsion due to coolant vaporization. The model developed for the study is that a real system can be described in terms of the expulsion of an equivalent coolant piston. Results of this study have indicated that expulsion times could be as short as several milliseconds. Expulsion times this short could cause high rates of reactivity introduction in a reactor with a positive sodium void coefficient.

Large melt-down facility

Although the foregoing in- and out-of-pile experiments permit the evaluation of certain phenomena and supply useful data about certain melt-down information, they do not permit one to observe directly the gross behavior of the reactor core during a complete melt-down. Fast-reactor safety has been evaluated on the basis of hypothetical accidents that involve complete melt-down of the core. The need to make conservative assumptions in these accidents regarding molten fuel motion, and the thermodynamic properties of the fuel at high temperatures and pressures, results in stringent requirements that might not be necessary if a better basis were found to establish the assumptions. The technical feasibility of conducting fast-reactor melt-down experiments to determine a more realistic maximum accident and to provide a better evaluation of fast-reactor safety has been investigated [68].

It was found that it is technically feasible to build a mock-up of a fast reactor that would contain a separate melt-down region of variable size in its center. Parametric experiments in which a portion of a core is melted could be conducted in this facility. Primary indication of events in the melt-down zone would be obtained from the observed reactivity changes and the calculated feedback of the driver core. No decision on the construction of this facility has been made.

CALCULATION OF ENERGY RELEASED FROM SUPER PROMPT CRITICAL ASSEMBLIES

The method of estimating the energy release, once the rate of reactivity has been determined by the hypothetical accident conditions, has been refined [1, 5, 69–76]. According to the basic Bethe-Tait method [46], the accident is assumed to proceed in the following manner. The reactivity is assumed added at a constant rate. This causes a power excursion to develop and the energy generated in this excursion produces pressures that accelerate the core material in the direction of the negative pressure gradient. The resulting disassembly of the core eventually terminates the excursion and determines the amount of energy generated.

Jankus [1] made the first major refinement in the method. He manipulated the equations analytically so that the solution for the energy release was reduced to a particularly simple numerical integration, thus eliminating an approximation that had been used to obtain numerical results. Jankus showed, within the assumptions of his method, that the energy release was a function of only two parameters.

During this same period, Stratton *et al.* [74] developed a computer program to determine the energy release in small solid assemblies like GODIVA or for situations in which the fuel configuration could be approximated by concentric shells of solid material spaced by gaseous material, equivalent to void in the calculation. This method gave good agreement with experimental energy bursts in GODIVA. The calculations with the spherical shells indicated that if there is a significant amount of heterogeneity a homogenized model may tend to over-estimate the energy release.

Subsequently, Los Alamos and Argonne collaborated on the development of the computer program AX-1 [77] for cases where the core is assumed to be a configuration of spherical shells of molten fuel materials. Reactivity changes are calculated by a modified S_n method rather than by the diffusion equation perturbation theory of the original Bethe-Tait calculation. Another important improvement is a complete treatment of the hydrodynamics equations, which had previously been approximated under the assumption that pressure is not influenced by the small material displacements during the power excursion. The studies using this code showed reasonable agreement with the earlier Jankus calculations for most conditions, but indicated that the added refinements can sometimes be important [70, 71, 75].

Nicholson [69] generalized the Bethe-Tait and Jankus techniques somewhat and verified the validity of the basic assumptions for a wide variety of conditions. He showed that a linear equation of state, as used in most earlier calculations, is often not suitable, and that even rather modest Doppler temperature coefficients of reactivity can have a major influence on the energy release. Using a digital computer program, he also applied the Bethe-Tait method to two-dimensional situations having finite cylindrical symmetry [69, 71, 78].

Wolfe *et al.* [72] studied large oxide cores and showed that the Doppler effect can limit the energy release and that with a strong negative Doppler coef-

ficient the energy release decreases with decreasing prompt neutron-generation time. In subsequent work [79], using an equation of state for uranium oxide at temperatures up to 25 000 °K [80], the same authors showed that much higher releases might result from oxide cores than from extrapolation of calculations on the metal-fueled Fermi core.

The general status of the energy release calculations is believed to be such that one can calculate reliable answers provided that: (*a*) the material configuration is reasonably uniform and given; (*b*) the rate of assembly is given; and (*c*) the Doppler effect and equation of state are known. The greatest uncertainties are in most cases due to the lack of knowledge about realistic fuel configurations and rates of assembly for real accidents, and uncertainties in the equation of state. The latter usually must be estimated theoretically or by extrapolation of low temperature data [81–84].

Since the energy release for a given set of conditions tends to be nominally proportional to the total heat capacity of the core, containment of melt-down accidents in large, dilute reactors may be quite difficult unless a large, negative Doppler effect is available to offset the effects due to greater size. In very large cores a complete melt-down appears remote but the possibility of a regional melt-down of relatively low energy release causing extremely rapid compression of the unmelted volume must be considered.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/284 Etats-Unis d'Amérique

Recherches sur la sûreté des réacteurs à neutrons rapides

par W. J. McCarthy *et al.*

A l'époque de la première Conférence de Genève, on se préoccupait du fait qu'il pouvait subsister certains phénomènes inexpliqués, voire inconnus, ce qui signifiait que la question de la sûreté des réacteurs à neutrons rapides n'était pas parfaitement élucidée. En particulier, on ignorait la cause de la fusion des éléments de combustible du réacteur EBR-I et on se demandait quel pouvait être l'ordre de grandeur de l'énergie libérée à la suite d'un accident déterminé. La situation est maintenant très différente; tous les aspects du comportement du réacteur EBR-I ont été prévus et la démonstration en a été faite. En réalité, on peut prévoir le comportement nucléaire de réacteurs à neutrons rapides des types EBR-II et Fermi si l'on peut décrire leur comportement physique. On peut même prévoir l'effet d'additions de réactivité mesurables, telles que mouvement des barres de contrôle, addition d'une matière hydrogénée ou effet de « cartouche froide », et on peut facilement en tenir compte dans les plans. Cependant, les modifications physiques que l'on ne peut pas bien décrire provoquent des effets de réactivité que l'on ne peut pas bien prévoir. Il est difficile de prévoir le comportement physique du mouvement de la matière et l'effet qu'il peut avoir sur la réactivité surtout lorsque le sodium entre en ébullition et que se manifestent des modifications dues à la cavitation dans le sodium et à la fusion du combustible. En pareil cas, il est difficile de prévoir certaines forces, gravitationnelles et autres. Les auteurs étudient la nature de ce genre de problèmes, la portée des programmes expérimentaux qu'il faudrait exécuter pour les résoudre, et d'autres programmes de recherche connexes relatifs à la sûreté.

Lorsqu'un canal est bouché ou que le circuit du caloporteur a perdu en partie sa capacité de dissiper la chaleur, l'accroissement de température du sodium qui en résulte pourrait provoquer localement l'ébullition du sodium, une perte de sodium, la fusion du combustible ou de la gaine et, par conséquent, un mouvement du combustible. On a étudié le comportement du combustible dans des incidents de ce genre, hors pile en utilisant des matières simulées, et en pile en faisant des expériences sur des aiguilles dans le réacteur TREAT. Ces expériences ont fourni des données fondamentales sur le comportement du combustible dans certaines circonstances anormales. On a envisagé de construire une installation pour procéder à des expériences à grande échelle sur la fusion des éléments de combustible. Une telle installation permettrait de provoquer la fusion de parties importantes du cœur d'un réacteur, afin d'obtenir une représentation plus complète de la cavitation dans le sodium, du mouvement du combustible et de leurs effets sur la réactivité.

On a procédé à un certain nombre d'études sur l'accident maximal, afin de déterminer la limite maximale théorique des risques potentiels que présente le réacteur à neutrons rapides. Dans le type d'accident analysé, il y a changement rapide de la disposition du combustible dans la situation extrêmement grave qui résulterait d'un concours de circonstances possible mais guère plausible. Dans les analyses les plus récentes, on tient compte de l'effet Doppler auquel on peut s'attendre. Les résultats obtenus indiquent que la libération d'énergie est très inférieure à celle que faisaient prévoir les analyses précédentes.

Bien que les petits réacteurs à neutrons rapides actuels aient des coefficients cavitaires du sodium négatifs, il a été établi que certains modèles de grands réacteurs à neutrons rapides pouvaient avoir des coefficients de température du sodium positifs. Afin de déterminer dans toute la mesure possible les différents aspects de ce problème, on a fait des études sur la stabilité des réacteurs à neutrons rapides ayant des coefficients de température du sodium positifs et des cœfficients de température de la structure négatifs. On a également étudié la saute de puissance qui pourrait se produire en cas d'expulsion du sodium d'un réacteur ayant un cœfficient cavitaire du sodium positif et un cœfficient Doppler négatif. Plus récemment, la Commission de l'énergie atomique a fait faire quatre études sur des réacteurs surgénérateurs à neutrons rapides de 1 000 MW(e), pour

déterminer dans quelle mesure les effets du sodium sur la réactivité peuvent causer des difficultés dans les grands réacteurs de ce genre.

Le programme de recherches sur la sûreté des réacteurs à neutrons rapides, dont l'exécution se poursuit aux Etats-Unis depuis quelques années, a permis de mieux connaître les problèmes, de perfectionner les méthodes utilisées pour les analyser, de résoudre qualitativement quelques-uns d'entre eux et de mieux en saisir la difficulté.

A/284 США

Исследования безопасности работы реактора на быстрых нейтронах

У. Дж. Маккарти *et al.*

На Первой женевской конференции делались предположения о возможности существования еще неизвестных явлений, связанных с безопасностью реакторов на быстрых нейтронах. Например, тогда не понимали причину расплавления активной зоны реактора EBR-I, а величина энергии, выделенной в результате этой аварии, была неизвестна. В настоящее время положение совершенно изменилось и поведение экспериментального реактора-размножителя (EBR-I) полностью изучено. Действительно, если физическое поведение реактора может быть описано, то ядерное поведение реакторов на быстрых нейтронах типа EBR-II и «Энрико Ферми» может быть предсказано. Эффекты введения реактивности, например, при перемещении регулирующих стержней, ввода водородсодержащего материала, или «холодного блока», могут быть предсказаны и легко компенсированы при проектировании. Однако физические изменения трудно поддаются описанию, приводят к изменениям реактивности, которые трудно правильно предсказать. Основные условия, при которых трудно предсказать физическое поведение, связаны с движением материала и последующим влиянием его на реактивность. Так, при кипении натрия изменяется реактивность вследствие образования пустот в натрии и раславления топлива. Поток, гравитационные и другие силы в таких условиях предсказать трудно. В докладе рассматриваются эксперименты, поставленные для решения этой задачи, и другие исследования по безопасности реакторов.

В аварийных условиях реактора с закупоренным каналом или с потерей теплосъемочной способности системы теплоносителя последующее повышение температуры натрия может вызвать местное его кипение, утечку натрия, расплавление тепловыделяющего элемента или оболочки и последующее перемещение топлива. Поведение топлива при таких авариях изучалось как вне реактора с помощью моделирующих средств, так и в самом реакторе на стержнях опытного реактора TREAT для изучения переходных процессов. Все эти опыты привели к пониманию поведения топлива при некоторых аварийных условиях. Была изучена практическая возможность создания установки для проведения опытов по расплавлению большой активной зоны. Такая установка позволит произвести расплавление большей части активной зоны реактора и даст более полную картину образования пустот в натрии, движения горючего и их влияния на реактивность.

Чтобы определить теоретически максимальную величину опасности реактора на быстрых нейтронах, были произведены исследования максимально возможной аварии. При такой аварии происходит быстрое перераспределение топлива при наиболее тяжелых условиях, являющихся последствием ряда возможных, но в то же время маловероятных обстоятельств. Последние исследования учитывают эффект Допплера. Полученные результаты показывают, что выделение энергии должно быть значительно меньше, чем предсказано ранее.

Хотя существующие небольшие реакторы на быстрых нейтронах имеют отрицательные пустотные коэффициенты реактивности, было обнаружено, что большие реакторы на быстрых нейтронах некоторых конструкций могут иметь положительные натриевые температурные коэффициенты. Для полной оценки этого вопроса изучалась стабильность реакторов на быстрых нейтронах, имеющих положительные натриевые температурные коэффициенты и отрицательные структурные температурные коэффициенты. Кроме того, изучался разгон реактора, который может произойти в результате выброса натрия из реактора при положительном пустотном коэффициенте и отрицательном допплеровском коэффициенте. Недавно Комиссия по атомной энергии провела четыре исследования на системах реакторов-размножителей на быстрых нейтронах мощностью 1000 *Мвт* (эл.) с целью определения условий, которые могут возникнуть в результате влияния пустот в натрии на реактивность больших реакторов.

Осуществление программы по изучению безопасности реакторов на быстрых нейтронах США позволило лучше понять основные задачи, улучшить методы анализа и дало качественное решение некоторых важных вопросов.

A/284 Estados Unidos de America

Las investigaciones sobre la seguridad de los reactores rápidos

por W. J. McCarthy et *al.*

Cuando se celebró la Primera Conferencia de Ginebra existía el temor de que hubiese fenómenos incom-

prendidos e incluso imprevistos, y de que no se controlaba bien la seguridad de los reactores rápidos. Por ejemplo, no se entendía la causa de la fusión del núcleo del EBR-I y se debatía la cantidad de energía liberada en un accidente determinado. Hoy la situación es muy distinta; se han predicho y demostrado todos los aspectos del comportamiento del EBR-I. De hecho, si se sabe describir el comportamiento físico del reactor, se puede predecir el comportamiento nuclear de los reactores rápidos de los tipos EBR-II y Fermi. Se pueden predecir los efectos de la presencia de un cartucho combustible frío o de las inserciones calculables de reactividad tales como el movimiento de las barras de control o la adición de materiales con hidrógeno, y tales efectos pueden controlarse fácilmente con un proyecto adecuado. Sin embargo, las alteraciones físicas que no se saben describir bien dan lugar a efectos de la reactividad que no se pueden predecir bien. El caso más importante en el que es difícil predecir el comportamiento físico del movimiento del material y el efecto consiguiente sobre la reactividad, es cuando tiene lugar la ebullición del sodio y hay cambios de reactividad a causa de la formación de cavidades en el sodio y de la fusión del combustible. En esta situación es difícil precedir las presiones dinámica, estática u otras. En la memoria se revisa la naturaleza de este tipo de problemas, el alcance de los programas experimentales para resolverlos y otros programas asociados de investigación.

En el caso del estado anormal de un reactor en el que haya un canal atascado o una pérdida en la capacidad de disipación del calor del sistema de transporte del mismo, el aumento consiguiente de la temperatura del sodio podría originar su ebullición localizada, la pérdida de sodio, la fusión del combustible o de la vaina y el correspondiente movimiento del combustible. Se ha estudiado el comportamiento del combustible en tales casos, fuera del reactor, haciendo uso de materiales simulados y, dentro del reactor, experimentando con barritas en el reactor TREAT. Estos experimentos han proporcionado los conocimientos básicos para interpretar el comportamiento del combustible en determinadas circunstancias anormales. Se ha estudiado la viabilidad de una instalación para llevar a cabo experimentos de fusión a gran escala. Una instalación de este tipo permitiría la fusión de fracciones grandes del núcleo de un reactor para obtener una idea más clara de la formación de cavidades en el sodio, del movimiento del combustible y de sus efectos sobre la reactividad.

Se han llevado a cabo algunos estudios del accidente máximo para determinar el límite teórico superior del riesgo potencial de los reactores rápidos. Se analiza un tipo de accidente en el que, en las condiciones más desfavorables, tiene lugar un reagrupamiento rápido del combustible a causa de una serie de circunstancias posibles pero probablemente inverosímiles. En los últimos análisis se tiene en cuenta el efecto Doppler que se puede prever. Los resultados obtenidos indican que la energía liberada es bastante menor que la predicha en análisis anteriores.

Aunque en los pequeños reactores rápidos existentes los coeficientes de temperatura del sodio sean negativos, se ha encontrado que en ciertos tipos de reactores rápidos grandes los coeficientes de temperatura del sodio podrían ser positivos. Para estimar de lleno el alcance de este problema, se ha estudiado la estabilidad de los reactores rápidos con coeficientes de temperatura del sodio positivos siendo negativos los de temperatura de la estructura. Además, se ha estudiado la excursión de potencia que podría ocurrir al expulsar el sodio de un reactor en el que el coeficiente de huecos del sodio fuese positivo y el coeficiente Doppler negativo. Más recientemente la AEC ha financiado cuatro estudios sobre reactores rápidos reproductores de 1 000 MW(e) en un esfuerzo para determinar el grado de dificultad real que pudiere resultar de los efectos del sodio sobre la reactividad en grandes reactores.

El programa de investigaciones sobre la seguridad de los reactores rápidos ha producido en los EE.UU. y en los años últimos un mejor entendimiento de los problemas, ciertos refinamientos en los métodos de análisis, soluciones cualitativas para algunos de ellos y un conocimiento mucho mejor de la complejidad de estos problemas.

Вопросы динамики водо-водяных реакторов при аварийном уменьшении циркуляции теплоносителя

Г. Н. Полетаев

1. ВВЕДЕНИЕ

Правильный выбор динамических характеристик ядерной энергетической установки позволит сделать ее максимально надежной в переходных и аварийных условиях.

Одним из необходимых разделов динамики реакторных систем водо-водяного типа является анализ переходных процессов при изменении циркуляции теплоносителя по первичному контуру. В этом случае представляют наибольший практический интерес следующие режимы работы установки:

а) аварийное уменьшение расхода теплоносителя;

б) увеличение расхода теплоносителя при запуске насоса.

Уменьшение расхода теплоносителя может привести к нежелательному резкому перегреву покрытий тепловыделяющих элементов даже после прекращения цепной реакции. Особенно опасно расплавление оболочки тепловыделяющего элемента в связи с возможностью выхода в контур продуктов деления. Кроме того, при температурах покрытия, несколько превышающих температуру плавления, опасно взаимодействие расплавленного металла с водой или паром с выделением значительного количества тепла [1]. При запуске насоса могут иметь место резкие неконтролируемые всплески нейтронной (и тепловой) мощности реактора при обычно имеющем место отрицательном температурном коэффициенте реактивности. Увеличение мощности в этом случае связано с высвобождением положительной реактивности при уменьшении средней температуры воды в реакторе (уменьшение температуры воды в реакторе вызывается как увеличением расхода теплоносителя, так и возможным забросом более холодной воды из трубопровода запускаемого насоса в реактор).

Необходимо найти соотношения между характерными параметрами процесса, приводящие к заранее заданному переходному режиму. Заданный переходный режим может быть охарактеризован предельно допустимым теплосодержанием теплоносителя в реакторе и отсутствием кризиса теплообмена (либо предельно допустимой температурой оболочки тепловыделяющего элемента). Математически задача сводится к системе обыкновенных дифференциальных уравнений, получаемых (частично) путем усреднения уравнений в частных производных:

а) уравнения движения теплоносителя;

б) уравнения передачи тепла;

в) уравнения нейтронной динамики;

г) уравнения для внешних воздействий.

Эти уравнения с теми или иными допущениями решаются, как правило, с использованием счетных машин.

Для быстрой части процесса может быть использована методика с частичным учетом распределенности параметров (по высоте реактора). Гидравлическая часть задачи (изменение циркуляции теплоносителя для однофазной жидкости) решается обычно независимо от тепловой. Нейтронная динамика при скачкообразном введении больших отрицательных реактивностей может быть также отделена от тепловой для быстрой части процесса.

Для целей предварительного проектирования, а также для лучшего понимания теплофизических и гидродинамических процессов, протекающих в реакторных установках, полезны также аналитические зависимости, позволяющие выявить в общем виде характерные критерии процесса и найти приближенно связь между ними, необходимую для ограничения переходного процесса допустимыми пределами изменения параметров.

Ниже приведены основные результаты работы и рекомендации, которых, на наш взгляд, целесообразно придерживаться при проектировании. Более подробные данные предполагается опубликовать в дальнейшем.

2. СИСТЕМА УРАВНЕНИЙ ДЛЯ ОПИСАНИЯ ИЗМЕНЕНИЯ ЦИРКУЛЯЦИИ ТЕПЛОНОСИТЕЛЯ ПО КОНТУРУ РЕАКТОРА

Уравнение движения теплоносителя легко может быть получено из уравнения механики

$$\frac{d}{dt}(m \cdot v) = \Sigma F \qquad (2.1)$$

и имеет следующий вид

$$\frac{\partial}{\partial t}\left(\frac{\gamma s w}{g}\right)+\frac{\partial}{\partial x}\left(\frac{\gamma s w^2}{g}\right)=F_l. \qquad (2.2)$$

Используя соответствующие аппроксимации для сил трения, давления, тяжести и местных сопротивлений, действующих в элементе dx контура, из выражения (2.2) можно получить уравнение для изменения во времени циркуляции несжимаемого теплоносителя по замкнутому контуру в следующем виде:

$$\psi\frac{dG}{dt}=\Delta p_{\text{дв}}-\Delta p_{\text{с}}, \qquad (2.3)$$

где

$$\psi=\frac{1}{g}\sum_j\frac{l_j}{s_j} \qquad (2.4)$$

(суммирование ведется по участкам контура с усредненными параметрами.) Дополняя выражение (2.3) известным уравнением для изменения угловой скорости вращения ротора насоса, получим следующую систему уравнений:

$$\left.\begin{aligned}\psi\frac{dG}{dt}&=\Delta p_{\text{дв}}-\Delta p_{\text{с}},\\ I\frac{d\omega}{dt}&=M_{\text{дв}}-M_{\text{с}}.\end{aligned}\right\} \qquad (2.5)$$

Систему (2.5) удобно записать в безразмерном виде

$$\left.\begin{aligned}\tau_{\text{г}}\frac{d}{dt}\left(\frac{G}{G_0}\right)&=\frac{\Delta p_{\text{дв}}}{\Delta p_0}-\frac{\Delta p_{\text{с}}}{\Delta p_0};\\ \tau_{\text{м}}\frac{d}{dt}\left(\frac{\omega}{\omega_0}\right)&=\frac{M_{\text{дв}}}{M_0}-\frac{M_{\text{с}}}{M_0},\end{aligned}\right\} \qquad (2.6)$$

где величины

$$\tau_{\text{г}}=\frac{\psi G_0}{\Delta p_0}=\left(\frac{1}{g}\sum_j\frac{l_j}{s_j}\right)\frac{G_0}{\Delta p_0}\simeq\frac{2E_{\text{т}}}{N_{\text{т}}}; \qquad (2.7)$$

$$\tau_{\text{м}}=\frac{I\omega_0}{M_0}=\frac{I\omega_0^2}{N_{\text{н}}}=\frac{2E_{\text{н}}}{N_{\text{н}}} \qquad (2.8)$$

имеют смысл характерных постоянных времени переходного процесса.

Величина $\tau_{\text{г}}$ — гидравлическая постоянная времени изменения циркуляции теплоносителя — характеризует отношение кинетической энергии, накопленной теплоносителем, к мощности, необходимой для преодоления сопротивления движению теплоносителя.

Величина $\tau_{\text{м}}$ — механическая постоянная времени изменения угловой скорости вращения ротора — характеризует отношение кинетической энергии, запасенной во вращающихся частях насоса, к мощности на валу двигателя насоса.

Из уравнения (2.2) легко получить аналогичную систему уравнений для реактора с разветвленным первичным контуром. В частности, для реактора с n петлями, в i-й петле которого может работать параллельно p_i насосов (для простоты с общими точками подключения), получим

$$\psi_{ik}\frac{dG_{ik}}{dt}+\psi_i\frac{dG_i}{dt}+\psi_{\text{р}}\frac{dG_{\text{р}}}{dt}=\Delta p_{\text{дв}_{ik}}-[\Delta p_{\text{с}_{ik}}+\Delta p_{\text{с}_i}+\Delta p_{\text{р}}], \qquad (2.9)$$

$$I_{ik}\frac{d\omega_{ik}}{dt}=M_{\text{дв}_{ik}}-M_{\text{с}_{ik}},$$

$$G_i=\sum_{k=1}^{p_i}G_{ik}\quad(k=1,2,\ldots,p_i),$$

$$G_{\text{р}}=\sum_{i=1}^{n}G_i\quad(i=1,2,\ldots,n).$$

Физически очевидными условиями совместности работы насосов в переходных процессах являются следующие:

а) равенство давлений в переходном процессе в общих точках соединения различных потоков теплоносителя;

б) непрерывность всех расходов и угловых скоростей.

Рассмотрим для иллюстрации условие совместности работы насосов в i-й петле, которое легко получается из (2.9) и имеет следующий вид:

$$\Delta p_{\text{дв}_{ik}}-\Delta p_{\text{с}_{ik}}-\psi_{ik}\frac{dG_{ik}}{dt}=\text{idem}\,(t). \qquad (2.10)$$

После обесточивания ik-го насоса условие (2.10) поддерживается автоматически за счет расходования «живой силы» потока $\left(\psi_{ik}\frac{dG_{ik}}{dt}\right)$ вплоть до достижения величины $G_{ik}=0$.

При запуске насоса условие (2.10) может быть выполнено лишь при развитии запускаемым насосом определенного напора. До этого момента можно считать, что насос работает в безрасходном режиме. Согласно (2.10) напор насоса должен быть не меньше, чем разность давлений на участке подключения.

Аналогичные с (2.10) условия будут иметь место и для совместной работы насосов в разных петлях.

Отметим, что при отсутствии захлопки на насосе, возможен обратный проток теплоносителя через остановленный насос. В этом случае уравнения должны быть изменены.

Заметим, наконец, что после обесточивания насоса инерция струи может стать внешней по отношению к колесу насоса силой. В этом случае насос окажется в режиме работы гидротурбины. Очевидно, что это будет иметь место в случае достижения следующего равенства:

$$\Delta p_{\text{дв}_{ik}}=0 \qquad (2.11)$$

(в общем случае, при $\omega_{ik}\neq 0$). С этого момента колесо насоса из движущей силы превращается в сопротивление [это должно быть заложено или

учтено в системе типа (2.9)]. С учетом вышесказанного возможно решение системы (2.9) на счетной машине. Аналитические решения могут быть получены с рядом упрощенных предположений. Ниже такие решения приводятся для интересующего нас случая обесточивания циркуляционных насосов.

3. ОБЕСТОЧИВАНИЕ ИЛИ ПОЛОМКА ЦИРКУЛЯЦИОННОГО НАСОСА

Под обесточиванием насоса будем понимать обесточивание питающих фидеров в случае насоса с электроприводом. Используя известные статические зависимости, можно приближенно написать

$$\Delta p_{\text{дв}} \simeq \frac{\Delta p_0}{1-\varphi_0^2}\left[\left(\frac{\omega}{\omega_0}\right)^2 - \varphi_0^2\left(\frac{G}{G_0}\right)^2\right], \qquad (3.1)$$

где величина φ_0, характеризующая крутизну характеристики насоса, может быть получена из расчета насоса или по опытным данным.

Характеристика сети обычно близка к квадратичной от расхода, то есть

$$\Delta p_{\text{с}} \simeq \Delta p_0 \left(\frac{G}{G_0}\right)^2. \qquad (3.2)$$

Момент сопротивления в насосе равен

$$M_{\text{с}} = M_{\text{г}} + M_{\text{тр}}, \qquad (3.3)$$

где

$$M_{\text{г}} \simeq \frac{\Delta p G}{\gamma \omega \eta_{\text{г}}^0 \eta_{\text{об}}^0}, \qquad (3.4)$$

$$M_{\text{тр}} \simeq M_{\text{тр}}^0 \left(\frac{\omega}{\omega_0}\right)^2. \qquad (3.5)$$

Для анализа удобно также следующее соотношение:

$$M_{\text{тр}} \simeq M_{\text{тр}}^0 \frac{\omega}{\omega_0}\frac{G}{G_0}, \qquad (3.6)$$

дающее весьма близкую к (3.5) зависимость в области $\frac{G}{G_0} \sim \frac{\omega}{\omega_0}$ (значение $M_{\text{тр}}$ в области $\frac{G}{G_0} \gg \frac{\omega}{\omega_0}$ несущественно). В первом приближении считаем, что с момента обесточивания насоса имеет место

$$M_{\text{дв}} = 0. \qquad (3.7)$$

Влияние $M_{\text{дв}}$ на процесс будет оценено отдельно.

Система (2.6) с учетом приведенных выше зависимостей запишется следующим образом:

$$\left.\begin{aligned} &(1-\varphi_0^2)\,\tau_{\text{г}}\frac{dx}{dt} = y^2 - x^2, \\ &(1-\varphi_0^2)\,\tau_{\text{м}}\frac{dy}{dt} = -xy\left[1-\varphi_0^2\left(\frac{M_{\text{тр}}^0}{M_0} + \right.\right. \\ &\left.\left. + \frac{M_{\text{г}}^0}{M_0}\frac{x^2}{y^2}\right)\right], \quad 0<\varphi_0<1, \end{aligned}\right\} \qquad (3.8)$$

где

$$x = \frac{G}{G_0}; \qquad (3.9)$$

$$y = \frac{\omega}{\omega_0}. \qquad (3.10)$$

Из (3.8) приближенно получим

$$x \simeq \frac{1}{1+\frac{t}{\tau_{\text{ц}}}}, \qquad (3.11)$$

где

$$\tau_{\text{ц}} = \tau_{\text{г}} + \tau_{\text{м}}. \qquad (3.12)$$

Величина $\tau_{\text{ц}}$ — постоянная времени спада циркуляции теплоносителя — оказывается в наиболее простом случае просто суммой определенных ранее частных постоянных времени.

Если $$N_{\text{т}} = N_{\text{н}} = N_{\text{пр}} \qquad (3.13)$$

(что может быть в идеальном случае, то есть при $\eta_{\text{г}} = \eta_{\text{об}} = 1$), то

$$\tau_{\text{ц}} = \frac{2(E_{\text{н}} + E_{\text{т}})}{N_{\text{пр}}}, \qquad (3.14)$$

то есть величина $\tau_{\text{ц}}$ характеризует отношение кинетической энергии, накопленной в контуре (теплоносителем и насосом), к мощности, затраченной на прокачку теплоносителя.

Точность выражения (3.11) тем больше, чем лучше выполняется соотношение

$$\frac{\tau_{\text{г}}}{\tau_{\text{м}}} + \frac{\tau_{\text{м}}}{\tau_{\text{г}}} + 2 \gg 1 - \varphi_0^2. \qquad (3.15)$$

Условие (3.15) не является жестким. Можно показать, что решение для (3.8) в области $\tau_{\text{г}} \sim \tau_{\text{м}}$ и $\varphi_0 = 0$ [отвечающей наихудшей точности приближения для (3.11)], имеет следующий вид:

$$x = y\sqrt{1-2\ln y}; \qquad (3.16)$$

$$\frac{t}{\tau} = \sqrt{\frac{2}{l}} \int_{\frac{1}{\sqrt{2}}}^{\sqrt{1-2\ln y}} e^{\eta^2} d\eta$$

или для близкого к данному случаю:

$$x = \frac{2\left(1+2\frac{t}{\tau}\right)}{\left(1+2\frac{t}{\tau}\right)^2+1} \quad \tau = \tau_{\text{м}} \quad \text{при} \quad \frac{\tau_{\text{г}}}{\tau_{\text{м}}} = \frac{1}{2};$$

$$y = \left[\frac{2}{\left(1+\frac{2t}{\tau}\right)^2+1}\right]^{\frac{\tau}{2\tau_{\text{м}}}} \quad \tau = \tau_{\text{г}} \quad \text{при} \quad \frac{\tau_{\text{г}}}{\tau_{\text{м}}} = 2. \qquad (3.17)$$

Сопоставление зависимостей (3.11), (3.16), (3.17), проведенное в безразмерных координа-

так $\left(x, \frac{t}{\tau}\right)$, показывает, что в области $\tau_г \sim \tau_м$ выражение (3.11) дает значительную относительную ошибку, особенно в области малых расходов. Однако интересующая нас абсолютная ошибка не превышает $\sim 0{,}05$ в долях x, что вполне приемлемо для практических расчетов. Используя зависимость (3.11), из системы (3.8) можно получить выражение для y, что позволяет определить момент перехода насоса в режим работы гидротурбины. Здесь представляют интерес следующие предельные случаи.

а) $\tau_м \gg \tau\tau$.

В этом случае (практически независимо от величины φ_0) получим

$$x \simeq y \simeq \frac{1}{1+\frac{t}{\tau_м}}, \qquad (3.18)$$

то есть процесс определяется инерцией вращающихся частей насоса. Это область пропорциональности между изменением угловой скорости вращения ротора насоса и расходом теплоносителя. При этом возможно использование применяемого многими авторами (например, [2]) соотношения $x = y$, что в применении к системе типа (2.9) может существенно сократить расчеты, а в некоторых случаях получить приближенные аналитические зависимости. При $\tau_м \gg \tau_г$ желательно оценить электромагнитное торможение ротора центробежного насоса с асинхронным двигателем закрытого типа, применяемым на водо-водяных установках высокого давления. После обесточивания насоса большой мощности возникает торможение ротора за счет взаимодействия спадающего вращающегося магнитного потока, сцепленного с ротором, с наведенными им в материале статора насоса вихревыми токами. Учитывая, что мощность потерь на вихревые токи $N_ф$ пропорциональна квадрату числа полных циклов перемагничивания (в нашем случае — квадрату угловой скорости вращения ротора) и квадрату наибольшей величины магнитной индукции $\left(\frac{R_р}{e^{\frac{L_р}{L_р}t}}\right)^2$, можно получить следующее выражение:

$$x \simeq \frac{e^{\eta_ф \frac{\tau_{эм}}{2\tau_ц}\frac{-2t}{\tau_{эм}}}}{e^{\eta_ф \frac{\tau_{эм}}{2\tau_ц}} + \frac{\tau_{эм}}{2\tau_ц}\int\limits_{\eta_ф \frac{\tau_{эм}}{2\tau_ц} e^{\frac{-2t}{\tau_{эм}}}}^{\eta_ф \frac{\tau_{эм}}{2\tau_ц}} \frac{e^{\xi}}{\xi} d\xi}, \qquad (3.19)$$

где

$$\eta_ф = \frac{N_ф}{N_н}; \qquad (3.20)$$

$$\tau_{эм} = \frac{L_R}{R_R} \qquad (3.21)$$

(L_R и R_R — соответственно индуктивность и активное сопротивление обмотки ротора). Для большинства практических случаев величина $\eta_ф \frac{\tau_{эм}}{2\tau_ц} \leqslant 1$. Учитывая, что в этом случае главной частью подынтегральной функции является величина $\frac{1}{\xi}$, из (3.19) получим

$$x \simeq \frac{1+\eta_ф \frac{\tau_{эм}}{2\tau_ц} e^{-\frac{2t}{\tau_{эм}}}}{1+\frac{t}{\tau_ц}+\eta_ф \frac{\tau_{эм}}{2\tau_ц}}. \qquad (3.22)$$

Таким образом, относительное уменьшение расхода в переходном режиме за счет дополнительного торможения ротора не превосходит в этом случае величины $\eta_ф \tau_{эм} / 2\tau_ц$, что обычно составляет величину не более 0,07 в долях от x. Из выражения (3.22) следует, что эти эффекты уменьшаются с увеличением величины $\tau_ц \simeq \tau_м$. При величине ($\tau_м \simeq 2 \div 3$) *сек* они могут не учитываться. Обычно величина $\tau_м \leqslant 1$ *сек*. Наличие у насоса дополнительных маховых масс может существенно увеличить величину $\tau_м$.

Заметим, что при $\tau_м \gg \tau_г$ объекта возможна приближенная оценка относительного спада расхода по экспериментально замеренному изменению относительной угловой скорости вращения ротора.

При $\tau_м \gg \tau_г$ стенда возможны измерения на ненатурном стенде. При этом следует иметь в виду, что согласно (2.8) для горячей и холодной систем имеет место (приближенно) соотношение

$$\tau_м^{гор} \simeq \frac{\gamma_{хол}}{\gamma_{гор}} \tau_м^{хол}, \qquad (3.23)$$

то есть в горячем контуре спад расхода теплоносителя будет более плавным.

Особо отметим, что для установок с $\tau_м \gg \tau_г$ опасной может быть механическая поломка или заклинивание ротора насоса. В этом случае возможен весьма резкий спад расхода теплоносителя, определяемый малой величиной $\tau_г$. Для таких установок в качестве предупредительной меры желательно иметь постоянно включенный (параллельно) резервный насос.

б) $\tau_г \gg \tau_м$.

В этом случае получим

$$x \simeq \frac{1}{1+\frac{t}{\tau_г}} \qquad t \leqslant t_0 \qquad (3.24)$$

(t_0 — момент перехода насоса в режим работы гидротурбины). Можно показать, что в этом случае величина y быстро стремится к следующему выражению:

$$y \simeq \frac{\varphi_0}{1+\frac{t}{\tau_г}} \qquad 0 < \varphi_0 < 1, \qquad (3.25)$$

то есть согласно (2.11) и (3.1) насос переходит в режим работы гидротурбины. С запасом можно принять, что с момента обесточивания насоса имеет место следующее соотношение:

$$x \simeq \frac{1}{1+\frac{t}{\tau_г^*}}, \tag{3.26}$$

где

$$\tau_г^* = \frac{\tau_г}{1+\frac{\Delta p_{\text{колеса}}}{\Delta p_0}}. \tag{3.27}$$

Величина $\frac{\Delta p}{\Delta p_0}$ колеса имеет порядок $0{,}3 \div 0{,}6$ и может быть получена при испытаниях насоса. (Отметим, что использование соотношения $x = y$ для упрощения выкладок в данном случае физически не оправдано. Однако оно может не привести к ошибке, если $\frac{\Delta p_{\text{колеса}}}{\Delta p_0} \ll 1$. Это объясняется тем, что вклад от вращающихся частей насоса при $\tau_г \gg \tau_м$ пренебрежимо мал и характер аппроксимации величины y для определения x несуществен.)

Экспериментальное определение спада расхода теплоносителя в этом случае возможно, по-видимому, лишь в реальном контуре. Заметим, что если испытания проводятся в холодной системе и номинальный режим автомоделен, то можно считать, что согласно (2.7) имеет место

$$\tau_г^{\text{гор}} \simeq \tau_г^{\text{хол}}, \tag{3.28}$$

то есть спад расхода теплоносителя будет примерно одинаков для горячей и холодной систем.

Анализ крупных отечественных стационарных реакторных установок (например, ВВЭР) показывает, что определяющей постоянной времени для рассматриваемого процесса является величина $\tau_г$. Качественно можно написать

$$\tau_г \simeq \frac{R}{w_0\left(\frac{\lambda}{d_г}+\frac{\Sigma\xi}{L}\right)}. \tag{3.29}$$

Средняя по контуру скорость теплоносителя (W_0) и насыщенность установки местными сопротивлениями на единицу длины ($\Sigma\xi/L$) мало меняются для различных водо-водяных установок. При этом обычно $\frac{\lambda}{d_г} \gtrsim \frac{\Sigma\xi}{L}$. Поэтому для установок с большим гидравлическим диаметром ($d_г$) величина $\tau_г$ относительно велика.

Для установок малогабаритных (например, у атомного ледокола «Ленин») процесс в основном определяется величиной $\tau_м$. Анализ величины $\tau_м$ затруднителен, так как зависимость $I(N_{\text{П}}, \omega_0)$ сугубо конструктивная. При проектировании насоса целесообразно поставить требование по возможности учитывать необходимость иметь величину $\tau_м$ достаточно большой $\gtrsim 1 \div 1{,}5$ *сек*, что возможно. Отметим, наконец, что обычно величины $\tau_г$ и $\tau_м$ имеют порядок $0{,}2 \div 1$ *сек*. (Последнее значение имеет место при отсутствии присоединенных маховых масс у насоса, например за счет выносного двигателя насоса для установки низкого давления.)

4. ТЕМПЕРАТУРНЫЙ РЕЖИМ РЕАКТОРА В НЕСТАЦИОНАРНЫХ УСЛОВИЯХ

Уравнения тепловой динамики реактора для процессов медленных или сравнимых со временем установления теплового равновесия в реакторе могут быть получены путем осреднения соответствующих уравнений в частных производных.

В наиболее простом виде они сводятся к обычным балансовым соотношениям, например, следующего вида:

$$\left.\begin{aligned} C_б\frac{dT_б}{dt} &= (q_0F)\frac{n}{n_0} - (kF)_б\,(T_б - T_{\text{п}}),\\ C_{\text{п}}\frac{dT_{\text{п}}}{dt} &= (kF)_б\,(T_б - T_{\text{п}}) - (kF)_{\text{п}}\,(T_{\text{п}} - T_т),\\ C_т\frac{dT_т}{dt} &= (kF)_{\text{п}}\,(T_{\text{п}} - T_т) - GC_т^{\text{уд}}(T_{\text{вых}} - T_{\text{вх}}), \end{aligned}\right\} \tag{4.1}$$

где

$$T_т = \varphi(T_{\text{вх}}, T_{\text{вых}}), \tag{4.2}$$

$$T_{\text{вх}} = f(t), \tag{4.3}$$

причем величина $T_т$ может быть задана из физических соображений.

Мы не будем рассматривать уравнения запаздывания и теплообмена в коммуникациях и парогенераторах, так как для интересующей нас в данной задаче быстрой части процесса режимы работы реактора и парогенераторов практически разделены во времени.

Система типа (4.1) совместно с приведенными ранее уравнениями вида (2.9) для изменения циркуляции теплоносителя и известными уравнениями нейтронной динамики позволяет описать (в точечном приближении) нестационарное поведение реактора.

Для выяснения чисто тепловых постоянных времени переходных процессов в реакторе рассмотрим сначала режим срабатывания аварийной защиты при постоянном расходе теплоносителя. Этот режим имеет много общего с задачей при переменном расходе, а также представляет самостоятельный интерес. При скачкообразном введении системой аварийной защиты большой отрицательной реактивности ($\rho_{\text{аз}} \gg \beta$) можно пренебречь температурными эффектами реактивности для быстрой части процесса (то есть при $T_{\text{вх}} \simeq \text{const}$.) Тогда при одной группе

запаздывающих нейтронов приближенно получим

$$\frac{n}{n_0} \simeq \frac{\beta}{\beta + \rho_{аз}} e^{-\frac{t}{\tau_{зап}}}. \qquad (4.4)$$

Можно показать, что влияние оболочки тепловыделяющих элементов на характер переходных процессов в водо-водяном реакторе мало ($C_п \ll C_б$; $C_п \ll C_т$). Величина теплоемкости покрытия всегда может быть эффективно учтена в других константах (например, в $C_б$). Тогда система (4.1) с учетом (4.4) может быть приведена к такому виду:

$$\frac{\tau_{пр}}{2}\tau_б \frac{d^2\theta}{dt^2} + \left(\tau_б + \frac{\tau_{tr}}{2}\right)\frac{d\theta}{dt} + \theta = \text{const}\frac{\beta}{\beta + \rho_{аз}} e^{-\frac{t}{\tau_{зап}}}, \qquad (4.5)$$

где $$\theta = \theta_т = \frac{T_т - T_{вх}^0}{T_т^0 - T_{вх}^0} \text{ при const} = 1; \qquad (4.6)$$

$$\theta = \theta_{q_1} = \frac{T_б - T_т}{T_б^0 - T_т^0} \text{ при const} = 1 - \frac{\tau_{пр}}{2\tau_{зап}}; \qquad (4.7)$$

$$\tau_б = \frac{C_б}{(kF)_б} = \frac{Q_б}{N_р}; \qquad (4.8)$$

$$\tau_{tr} = \frac{C_т + C_б}{GC^{уд}} = \frac{2Q}{N_р}; \qquad (4.9)$$

$$\tau_{пр} = \frac{C_т^т}{GC_т^{уд}} = \frac{L}{w}. \qquad (4.10)$$

Величина $\tau_б$ — постоянная времени изменения температуры тепловыделяющего элемента (блока) — характеризует отношение избыточного (по сравнению с теплоносителем) тепла, аккумулированного в блоке, к мощности реактора.

Величина τ_{tr} — транспортная, или конвективная, постоянная времени реактора — характеризует отношение тепла, высвобождаемого тепловыделяющими элементами и теплоносителем при расхолаживании их от средней температуры теплоносителя до температуры его на входе в реактор, к мощности реактора.

Так как имеет место соотношение

$$\tau_{tr} = \tau_{пр}\left(1 + \frac{C_б}{C_т}\right), \qquad (4.11)$$

то можно сказать, что величина τ_{tr} характеризует также запаздывание в конвективном переносе тепла (по длине реактора) теплоносителем за счет конечной теплоемкости блока, то есть является временем распространения теплового импульса по реактору. Отметим также, что в рассматриваемом здесь энергетическом содержании постоянных времени реактора величина

$$\tau_р = \tau_б + \frac{\tau_{tr}}{2} = \frac{Q_{ак}}{N_р}\theta \qquad (4.12)$$

есть постоянная времени установления теплового равновесия в реакторе и характеризует отношение запасенного в реакторе тепла к его мощности.

Уравнение (4.5) может быть легко решено. Нас будут интересовать некоторые предельные случаи, имеющие практический интерес:

а) $\tau_{tr} \gg \tau_б$.

Тогда из (4.5) приближенно получим ($\tau_{зап} \to \infty$)

$$\theta_т \simeq \frac{\beta}{\beta + \rho_{аз}} + \frac{\rho_{аз}}{\beta + \rho_{аз}} e^{-\frac{2t}{\tau_{tr}}}, \qquad (4.13)$$

то есть переходный процесс определяется временем распространения тепловой волны по реактору. Физически это означает, что избыточным теплом в тепловыделяющих элементах по сравнению с теплоносителем пренебрегается. В этом случае температура элемента близка к температуре теплоносителя, и можно говорить о расхолаживании (разогреве) реактора в целом.

б) $\tau_б \gg \tau_{tr}$.

В этом случае из (4.5) будем иметь

$$\theta_т \simeq \theta_{q_l} \simeq \frac{\beta}{\beta + \rho_{аз}} + \frac{\rho_{аз}}{\beta + \rho_{аз}} e^{-\frac{t}{\tau_б}}, \qquad (4.14)$$

то есть переходный процесс определяется тепловой инерцией тепловыделяющих элементов. При этом относительный спад температуры теплоносителя примерно совпадает с относительным спадом теплового потока. Физически это означает, что теплоемкостью теплоносителя в этом случае можно пренебречь.

Из рассмотренного выше следует, что режим обесточивания насосов более опасен для реактора с элементами с высоким тепловым потенциалом, то есть с $\tau_б \gg \tau_{tr}$ (что, например, имеет место на тепловыделяющих элементах из спеченной двуокиси урана).

Можно написать, что

$$\tau_б \simeq \frac{d_б^2}{Aa_б} + \frac{C_б}{\alpha_{заз}^{эф} F_б} + \frac{C_б}{\alpha_п F_б} + \frac{C_б}{\alpha_т F_б}, \qquad (4.15)$$

где

$$d_б = \frac{4F_б}{П_б} \qquad (4.16)$$

(A — коэффициент формы блока — может быть получен из решения уравнения теплопроводности в стационарных условиях).

Для блоков из двуокиси урана средних размеров (например, как в реакторах атомного ледокола «Ленин») основными будут первые два члена в выражении для $\tau_б$ (за счет малой теплопроводности двуокиси урана и газового зазора).

Для блоков из двуокиси урана больших размеров величина $\tau_б \sim \frac{d_б^2}{Aa_б}$. Обычно $\tau_б \sim 2 \div 5$ *сек* для элементов из двуокиси урана.

Легко показать, что при выполнении условия

$$\frac{\tau_{тэ}}{\tau_{ц}} \leqslant \frac{\rho_{аз}}{\beta + \rho_{аз}} \qquad (4.17)$$

не произойдет повышения температуры теплоносителя в реакторе непосредственно после обесточивания насосов с одновременным срабатыванием аварийной защиты. Здесь

$$\tau_{тэ} = \frac{d_{б}^2}{A a_{б}} + \frac{C_{б}}{\alpha_{заз}^{эф} F_{б}} + \frac{C_{б}}{\alpha_{п} F_{б}}. \qquad (4.18)$$

Физически условие (4.17) весьма простое: оно требует, чтобы начальный спад теплового потока был резче, чем спад расхода теплоносителя. Для глубокого сброса мощности ($\rho_{аз} \gg \beta$) условие (4.17) примет следующий вид:

$$\tau_{тэ} \leqslant \tau_{ц}. \qquad (4.19)$$

Как указывалось выше, величина $\tau_{ц} \sim 1 \div 1{,}5$ *сек* для установок, где на насосе не имеется дополнительных маховых масс. Поэтому условие (4.19) на блоках из двуокиси урана может существенно не выполняться. В этом случае следует ожидать значительного повышения температуры теплоносителя, особенно при имеющем место запаздывании между поступлением сигнала в систему аварийной защиты и началом ее эффективного влияния на изменение мощности реактора ($\tau_{аз}$ порядка нескольких десятых долей секунды).

Приближенный учет распределенности параметров (по высоте реактора) может быть сделан следующим образом.

Полагая, что в уравнении теплообмена для движущейся жидкости

$$G \frac{\partial i_{г}}{\partial x} + \gamma s \frac{\partial i_{т}}{\partial t} = q_{l} \qquad (4.20)$$

переменные в выражении для q_l разделяются, можно написать его решение в таком виде:

а) для частиц теплоносителя, находившихся в момент начала переходного процесса ($t' = 0$) внутри активной зоны реактора ($t_0' \leqslant 0$),

$$i_{т}(x, t) = i_{стац}\left(x - \int_0^t w(t'')\,dt'', 0\right) + \int_{x - \int_0^t w(t'')dt''}^{x} q_{l}(x') \frac{N(t')}{N_0} \frac{dx'}{G(t')}, \qquad (4.21)$$

где

$$x' = x - \int_{t'}^{t} w(t'')\,dt'', \qquad (4.22)$$

$$i_{стац}\left(x - \int_0^t w(t'')\,dt'', 0\right) = \frac{1}{G_0} \int_{0}^{x - \int_0^t w(t'')dt''} q_{l}(x'')\,dx''; \qquad (4.23)$$

б) для частиц теплоносителя, находившихся в момент начала переходного процесса вне активной зоны реактора ($t_0' \geqslant 0$),

$$i_{т}(x, t) = i_{вх}(0, t_0') + \int_0^x q_{l}(x') \frac{N(t')}{N_0} \frac{dx'}{G(t')} \qquad (4.24)$$

[момент t_0' входа рассматриваемой в точке (x, t) частицы теплоносителя в активную зону реактора определяется из выражения (4.22) при $x' = 0$]. Относительная величина теплового потока, передаваемого теплоносителю

$$\frac{N}{N_0} = \frac{(kF)_{п}(T_{п} - T_{т})}{(kF)_{п}^{0}(T_{п}^{0} - T_{т}^{0})}, \qquad (4.25)$$

может быть определена из задачи в точечном приближении. Заметим, что для тепловыделяющих элементов с низким тепловым потенциалом ($\tau_{б} \ll \tau_{tr}$) возможно существенное упрощение задачи, а именно: в зависимостях (4.21) — (4.24) достаточно заменить скорость теплоносителя на скорость распространения тепловой волны

$$w_{tr} = w \frac{C_{т}}{C_{т} + C_{б}}. \qquad (4.26)$$

При этом вместо неизвестного теплового потока можно приближенно использовать относительный спад нейтронной мощности.

В рассмотренном выше плане возможно решение задачи как в точечном приближении, так и с учетом распределенности параметров. Однако эти решения громоздки. Для предварительных (с запасом) оценок можно использовать точечную модель в наиболее грубой форме, полагая, что

$$i_{вых} \simeq i_{вх} + \frac{N}{G}. \qquad (4.27)$$

Можно показать, что для элементов с $\tau_{б} \gg \tau_{tr}$, для которых данная задача актуальна, спад теплового потока при уменьшении расхода теплоносителя вплоть до расхода малой (установившейся) циркуляции практически совпадает с таковым при постоянном расходе. Это связано с тем, что уменьшение теплового потока за счет роста термического сопротивления между теплоносителем и блоком в данном случае мало. С учетом сказанного легко получить связь характерных параметров процесса, при которой относительная температура (или энтальпия) теплоносителя не превысит заданной величины $\theta_{пред.}$, например, в следующем виде:

$$e^{\frac{\tau_{ц}}{\tau_{б}} \frac{1-\varkappa}{\varkappa}} \leqslant e^{-\frac{\tau_{аз}}{\tau_{б}}} \left[\left(1 + \frac{\rho_{аз}}{\beta}\right) \varkappa \theta_{пред} - \frac{1}{\rho_{аз}/\beta}\right], \qquad (4.28)$$

где

$$\varkappa = \frac{G_{уст}}{G_0}; \qquad (4.29)$$

$$\theta_{\text{пред}} = \frac{i_{\text{пред}} - i^0_{\text{вх}}}{i^0_{\text{вых}} - i^0_{\text{вх}}}. \qquad (4.30)$$

Величину $\theta_{\text{пред}}$ целесообразно, по-видимому, задать из условия отсутствия кипения теплоносителя в реакторе. Величина $\rho_{\text{аз}}/\beta$, выбираемая из условия перекрытия температурного эффекта при расхолаживании реактора до величины $\sim T^0_{\text{вх}}$, составляет $2 \div 4$. В этом диапазоне она не лимитирует в рассматриваемом процессе. Влияние $\rho_{\text{аз}}/\beta$ уменьшается с ростом величины $\tau_{\text{б}}$, так как в этом случае процесс все более определяется избыточным теплом тепловыделяющих элементов, а не внутренним тепловыделением.

Для тепловыделяющих элементов с $\tau_{\text{б}} \ll \tau_{\text{tr}}$ проблема аварийного теплоотвода после обесточивания насосов практически снимается. В этом случае условие (4.17) для отсутствия всплеска температуры теплоносителя выполняется. Для установки такого типа в основном существенны два момента:

а) величина $\tau_{\text{аз}}$.

С запасом можно считать, что для ограничения выбега температуры теплоносителя за время $\sim\tau_{\text{аз}}$ (при примерно номинальной мощности реактора) величиной $\theta_{\text{пред}}$ необходимо

$$\frac{\tau_{\text{аз}}}{\tau_{\text{ц}}} < \theta_{\text{пред}} - 1. \qquad (4.31)$$

Для напряженных энергетических водо-водяных реакторов величина $\theta_{\text{пред}}$, выбираемая из условия отсутствия кипения теплоносителя, составляет $2 \div 3$, и условие (4.31) может быть легко выполнено (при отсутствии механической поломки в насосе).

Заметим попутно, что желательно также иметь

$$\tau_{\text{аз}} \lesssim \tau_{\text{пр}}. \qquad (4.32)$$

В этом случае интеграл тепла, набираемый частицей теплоносителя за время $\tau_{\text{аз}}$, будет мал;

б) для обеспечения заданной $\theta_{\text{пред}}$ в конце процесса необходимо (с запасом)

$$\varkappa\theta_{\text{пред}} \geqslant \frac{\beta}{\beta + \rho_{\text{аз}}}. \qquad (4.33)$$

Заметим, наконец, что приближение типа (4.27) может быть существенно улучшено выбором физически оправданного эффективного центра теплоотвода в реакторе и приведено, например, к следующему виду:

$$i_{\text{вых}}(t) \simeq i_{\text{вх}}\left(t - \frac{\tau_{\text{пр}}(t)}{2}\right) + \frac{N\left(t - \frac{\tau_{\text{пр}}(t)}{2}\right)}{G\left(t - \frac{\tau_{\text{пр}}(t)}{2}\right)}. \qquad (4.34)$$

Выражение типа (4.34) может дать близкое совпадение с задачей с распределенными (по высоте реактора) параметрами.

5. КРИЗИС ТЕПЛООБМЕНА В РЕАКТОРЕ В ПЕРЕХОДНОМ ПРОЦЕССЕ

Используя подходящие статические зависимости для определения критических тепловых нагрузок, можно оценить запас по кризису теплообмена в нестационарных условиях. Например, используя (при постоянном давлении) зависимость [3]

$$q_{\text{кр}} = \text{const}\,(w\gamma)^{1/3}(T_s - T_{\text{т}})^{1/3} \qquad (5.1)$$

можно показать, что для рассматриваемой нами задачи запас по критической нагрузке в переходном процессе будет сразу уменьшаться, если

$$\frac{\tau_{\text{б}}}{\tau_{\text{ц}}} \gtrsim 2\frac{\rho_{\text{аз}}}{\beta + \rho_{\text{аз}}}. \qquad (5.2)$$

При этом величина малой (установившейся) циркуляции может определить лишь тот минимум, до которого этот запас упадет. Отсюда следует, что при выполнении (5.2) должно быть

$$n_{q_{\text{кр}}}(x, 0) = \frac{1}{k_{\text{нест}}} \geqslant 1, \qquad (5.3)$$

то есть уже в стационарном режиме необходимо иметь динамический запас. Отметим, что переход в кипящий режим потребует дальнейшего увеличения этого запаса. Из ранее приведенных цифр легко видеть, что соотношение (5.2) будет, например, иметь место на тепловыделяющих элементах из двуокиси урана. Предварительная (с запасом) оценка режима может быть сделана в точечном приближении. Предположим, что:

а) положение опасной точки в реакторе ($x_{\text{оп}}$) не меняется в переходном режиме,

б) минимум запаса по критической нагрузке ($n_{q\,\text{кр}}$) имеет место в момент выхода на установившуюся (малую) циркуляцию. (Эти допущения достаточно хорошо согласуются с более точными расчетами с учетом распределенности параметров вдоль реактора.) В этом случае [с учетом (5.1)] можно получить следующую связь параметров, не приводящую (с запасом) к кризису теплообмена в переходном режиме, связанном с обесточиванием насосов и срабатыванием аварийной защиты:

$$\varkappa \leqslant \frac{\left[1 - \frac{\theta_{\text{пред}} - 1}{\theta_s(x_{\text{оп}}) - 1}\right]^{1/3}}{\theta^2_{\text{пред}}\, k^2_{\text{нест}}}, \qquad (5.4)$$

где

$$\theta_s = (x_{\text{оп}}) = \frac{T_s - T^0_{\text{вх}}}{T^0_{\text{т}}(x_{\text{оп}}) - T^0_{\text{вх}}} \qquad (5.5)$$

причем величина $\varkappa$ в выражении (5.4) выбирается из условия (4.28). Зависимости типа (5.4) могут быть получены и при отличных от (5.1) аппроксимациях для определения $q_{\text{кр}}$. При необ-

ходимости частых оценок зависимости (4.28) и (5.4) могут быть изображены графически в виде номограмм.

При невыполнении условия типа (5.4) возможен кризис теплообмена, несмотря на прекращение цепной реакции. Высокая тепловая нагрузка в реакторе является в этом случае весьма специфичной, так как обусловлена в основном большим температурным потенциалом тепловыделяющих элементов. В первые моменты времени после образования паровой пленки на поверхности блока происходит выравнивание температур по сечению сердечника и оболочки элемента. Дальнейшее расхолаживание (разогрев) блока в целом зависит от соотношения температурного напора тепловыделяющего элемента, его внутреннего тепловыделения и теплоотвода после кризиса теплообмена. Приближенное аналитическое решение задачи может быть получено из системы типа (4.1) в предположении, что температура теплоносителя постоянна, а коэффициент теплоотдачи к теплоносителю в момент кризиса теплообмена меняется скачкообразно до некоторой постоянной величины $d_т^{пл}$. Тогда для наиболее опасного с точки зрения расплавления блока случая ($C_п \ll C_б$) получим для не слишком малых t:

$$\theta_п \simeq \frac{n^*_{вн}}{n_{пл}}\left[1 - \left(1 - \theta^*_б \frac{n_{кр}}{n^*_{вн}}\right) e^{\frac{t}{\tau_б^{пл}}}\right]; \qquad (5.6)$$

где

$$\theta_п = \frac{T_п - T_т^0}{T_п^0 - T_т^0}; \qquad (5.7)$$

$$\theta^*_б = \frac{T^*_б - T^*_п}{T^0_б - T^0_п} = \frac{T^*_б - T^0_т}{T^0_п - T^0_т}\,\frac{\tau_1}{\tau_1 + \tau_2^*}; \qquad (5.8)$$

$$\tau_1 = \frac{d_б^2}{A a_б} + \frac{C_б}{\alpha^{эф}_{заз} F_б} + \frac{C_б}{2a_п F_б}; \qquad (5.9)$$

$$\tau_2 = \frac{C_б}{2\alpha_п F_б} + \frac{C_б}{\alpha_т F_б}; \qquad (5.10)$$

$$\tau_2^{пл} = \frac{C_б}{2\alpha_п F_б} + \frac{C_б}{\alpha_т^{пл} F_б}; \qquad (5.11)$$

$$\tau_б^{пл} = \tau_1 + \tau_2^{пл}; \qquad (5.12)$$

$$n^*_{вн} = \frac{\beta}{\beta + \rho_{аз}} e^{\frac{t'^*}{\tau_{зап}}}; \qquad (5.12а)$$

$$n_{пл} = \frac{\tau_2}{\tau_2^{пл}}; \qquad (5.13)$$

$$n_{кр} = \frac{\tau_1 + \tau_2^*}{\tau_1 + \tau_2^{пл}} \qquad (5.14)$$

(* относится к моменту времени, непосредственно предшествующему кризису теплообмена). Из (5.6) получим, что начальный всплеск величины $\theta_п$ составляет следующую величину:

$$\theta_п(0) \simeq \frac{\tau_2^{пл}(\tau_1 + \tau_2^*)}{\tau_2(\tau_1^* + \tau_2^{пл})}\theta^*_б. \qquad (5.15)$$

При полном прекращении теплоотвода ($\tau_2^{пл} \to \infty$) из (5.15) получим

$$\theta_п(0) \simeq \frac{\tau_1 + \tau_2^*}{\tau_2} = \frac{T^*_б - T^0_т}{T^0_п - T^0_т}, \qquad (5.16)$$

что означает, что $T_п \approx T_б^*$, то есть температура оболочки не превысит в период выравнивания температур по блоку средней температуры сердечника, имевшей место в момент кризиса теплообмена, что физически очевидно. Дальнейшее поведение $T_п$ определяется, как это видно из (5.6), соотношением

$$\theta^*_б \frac{n_{кр}}{n^*_{вн}} \gtrless 1. \qquad (5.17)$$

Например, для понижения температуры оболочки после кратковременного всплеска для наиболее опасного (теоретически) случая — кризиса теплообмена в момент срабатывания аварийной защиты — необходимо

$$n_{кр} > \frac{\beta}{\beta + \rho_{аз}}. \qquad (5.18)$$

Анализ реальных величин $d_т^{пл}$ показывает, что для элементов с $\tau_б \gg \tau_{tr}$ (например, из двуокиси урана) условие (5.18) может быть выполнено. Это означает, что температура оболочки в переходном режиме не превысит (с запасом) средней температуры сердечника, имевшей место в номинальном режиме. Следовательно, если

$$T_п^{плавл} \gtrsim T^0_б, \qquad (5.19)$$

то расплавления оболочки для тепловыделяющих элементов этого типа может не произойти. Условие (5.19), как правило, выполняется на элементах из двуокиси урана для тугоплавких материалов оболочки (сталь, циркониевый сплав) при достаточно высоких тепловых нагрузках. Для тепловыделяющих элементов этого типа с покрытием из алюминиевого сплава необходимы существенно более низкие тепловые нагрузки.

Если $\tau_1 \ll \tau_2$, то после выравнивания температур по блоку возможен дальнейший рост величины $T_п$. При этом необходимо

$$\theta_п^{плавл} > \frac{n^*_{вн}}{n_{пл}}. \qquad (5.20)$$

Аналогичное условие должно иметь место и для температуры сердечника тепловыделяющего элемента в этом случае.

Таким образом, в случае кризиса теплообмена на недогретой воде возможны условия, не приводящие к расплавлению оболочки или сердечника тепловыделяющего элемента. Конкретный анализ при проектировании может определить допустимость кратковременной работы реактора в области повышенных температур тепловыделяющих элементов.

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ОСНОВНЫЕ ОБОЗНАЧЕНИЯ

G — весовой расход теплоносителя
ω — угловая скорость вращения
J — момент инерции
M — момент вращения
m — масса
Δp — перепад давлений
N — мощность
E — кинетическая энергия
Q — количество тепла
w — скорость теплоносителя
S — проходное сечение для теплоносителя
F — поверхность теплообмена
d — диаметр
Π — периметр
l, L — длина
ξ — коэффициент местного сопротивления
η — коэффициент полезного действия
n — поток нейтронов
ρ — реактивность
β — доля запаздывающих нейтронов
i — теплосодержание
T — температура
C — теплоемкость
γ — удельный вес
α — коэффициент теплоотдачи
K — коэффициент теплопередачи
a — коэффициент температуропроводности
q — тепловой поток
g — ускорение силы тяжести
τ — постоянная времени
t — время

ИНДЕКСЫ

0 — начальный режим
р — реактор
с — сопротивление
г — гидравлический
дв — движущий
м — механический
об — объемный
тр — трение
т — теплоноситель
н — насос
эм — электромагнитный
l — относится к единице длины
уст — установившийся
s — насыщение
б — блок
ф — Фуко
п — покрытие
вх — вход
вых — выход
кр — кризис
пл — пленочный
вн — внутренний
заз — зазор
плавл — плавление
пр — проход
пред — предельный
зап — запаздывание
стац — стационарный
ц — циркуляция
тэ — тепловыделяющий элемент
гор — горячий
хол — холодный
пр — прокачка
аз — аварийная защита
уд — удельный
ак — аккумулированный
эф — эффективный

ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/382 USSR

Dynamics of water-moderated water-cooled reactors on accidental drop in coolant circulation

By G. N. Poletaev

In the first part of the paper, a system of equations is developed for computing the changes that occur in the coolant circulation in the primary circuit of a multi-loop, water-moderated, water-cooled reactor when the operating conditions of the circulating pumps are disturbed or the pumps fail. The conditions of system consistency are cited, and analytical solutions are given for a number of cases.

A detailed analysis is made of the effects of electrical or mechanical failure of the circulating pumps. Typical time constants for the drop in circulation are worked out and analysed for different reactors with various types of circulating pump.

The analytical solutions are compared with the corresponding computer calculations and experimental data. Bench tests for determining the drop in coolant circulation are strongly recommended.

The second part of the paper comprises an analysis of the conditions that arise in a water-moderated, water-cooled reactor when the power supply to the circulating pumps fails. Reactors with various types of fuel element are considered. Criteria are established which will enable the operating staff to keep thermal conditions in the reactor within the specified limits (e.g., lack of bulk boiling of coolant and lack of heat-exchange crisis, or maximum permissible temperature of the fuel element in the event of heat exchange crisis). Some calculated data are quoted and compared with the experimental data available.

A/382 URSS

Problèmes de la dynamique des réacteurs ralentis et refroidis à l'eau dans le cas d'une baisse fortuite de la circulation du réfrigérant

par G. N. Poletaev

La première partie du mémoire développe un système d'équations décrivant les variations que subit la circulation du réfrigérant dans le circuit primaire d'un réacteur à boucles multiples modéré et refroidi à l'eau dans le cas d'un changement de régime ou d'un arrêt du fonctionnement des pompes de circulation. On donne les conditions de la cohérence du système. On indique des solutions analytiques pour certains cas.

On étudie en détail les effets d'une panne électrique ou d'une détérioration des pompes de circulation. On établit les constantes de temps caractéristiques de la chute de la circulation et on analyse ces constantes pour différentes installations avec des pompes de circulation de types différents.

On compare les solutions analytiques avec les calculs faits sur une machine à calculer et avec les données expérimentales. On donne des recommandations pour une expérience au banc d'essais visant à déterminer la chute de la circulation du réfrigérant.

La deuxième partie du mémoire est consacrée à l'analyse du régime thermique dans le réacteur en cas de panne de l'alimentation électrique des pompes. On envisage des réacteurs avec différents types d'éléments combustibles. On établit des critères pour que le régime thermique du réacteur demeure dans des limites données (par exemple absence d'ébullition en masse du réfrigérant et de crise d'échange de chaleur, ou température maximale admissible de la gaine de l'élément combustible dans le cas de crise d'échange de chaleur). On donne certains résultats de calculs, qui sont comparés aux données expérimentales disponibles.

A/382 URSS

Problemas dinámicos de los reactores moderados y refrigerados por agua, en caso de disminuir por avería la circulación del refrigerante

por G. N. Poletaev

En la primera parte de la memoria se establece un sistema de ecuaciones que describe la variación de la circulación del refrigerante en el circuito primario de un reactor, con varios circuitos, refrigerado y moderado por agua, al pertubarse el régimen de trabajo de las bombas de circulación. Se dan las condiciones de compatibilidad del sistema. Se presentan soluciones analíticas para algunos casos.

Se estudia con detalle el caso de rotura o de quedar sin corriente las bombas de circulación, Se determinan las constantes de tiempo características de la caída de circulación y se analizan para diferentes reactores con bombas de circulación de tipos diversos.

Se comparan las soluciones analíticas con los resultados de cálculos realizados con máquinas calculadoras, y con datos experimentales. Se hacen recomendaciones para el montaje de un experimento de prueba con el fin de determinar la caída de circulación del refrigerador.

En la segunda parte de la memoria se analiza el régimen térmico en el reactor refrigerado y moderado por agua, al quedar las bombas sin corriente. Se consideran reactores con diferentes tipos de elementos combustibles. Se fijan criterios que permiten al personal encargado de la operación mantener las condiciones térmicas en el reactor dentro de límites dados (p. ej., evitar la ebullición volumétrica del refrigerante y el fallo del intercambio de calor, o limitar la temperatura de las vainas de los elementos combustibles en caso de disminuir el intercambio de calor). Se presentan algunos resultados númericos que se comparan con los datos experimentales existentes.

On some dynamic properties of a power station with a heavy-water gas-cooled reactor

By P. Štirský * and J. Hůlovec **

The first Czechoslovak nuclear power station is of the all-condensation type, with one gas-cooled heavy-water reactor using natural uranium, having a heat output of 600 MW(th), feeding three blocks. Each block contains: (*a*) two one-stage radial CO_2 blowers at 60 bars, each designed for 278 kg/s with 12 bars over-pressure; (*b*) two steam generators of the tube-in-tube type, generating medium- and low-pressure steam: the output of each is 23.9 kg steam/s, 31.5 bars, 400 °C in the medium-pressure part, and 10.4 kg steam/s, 2.2 bars, 185 °C in the low-pressure part; (*c*) one turbo-generator, 50 MW (e). A detailed description of the power station is given in [1 and 5].

The use of a gas-cooled heavy-water reactor leads to several specific properties of the design of the power station, which influence its dynamic properties:

(*a*) A relatively low uranium charge, high heat transfer coefficient and large transfer areas of the fuel element, cause short time constants of the gas temperature and uranium temperature for variations of the neutron output of the reactor and coolant flow-rate.

(*b*) The high temperature gradient of the reactor causes large temperature variations of the gas at the reactor outlet, and variations of the uranium temperature with changes of the neutron flux and gas flow-rate.

(*c*) The moderator, thermally insulated from the cooling gas, does not allow the favourable influence of the reactor's self-regulation to be utilised through the heat coefficient of reactivity of the moderator.

(*d*) The time constants of the steam generators for the two types of steam are longer by one to two orders of magnitude with respect ot the time constants of the reactor.

(*e*) The exit temperature from the steam generator varies very little with different output levels, even in the case of transition regimes. This practically ensures the self-sufficiency of the reactor control system in the case of disturbances in the steam generators due to changes in the reactor input temperature.

(*f*) The requirement of a minimum proportion of pumping power, and stability of steam parameters require changes of the reactor output level and the output of the whole power station to be achieved by varying the coolant flow-rate, the temperature gradient of the reactor being nearly constant.

The strict operational demand, that a certain permitted maximum temperature of the fuel cans may not be exceeded under any operational as well as breakdown conditions, makes great demands on the reliability, precision and quality of control loops. The maximum uranium temperature, however, cannot be used directly as a regulator and the can temperature is kept within acceptable limits by the control of the gas output temperature of the reactor and by means of a knowledge of the dynamic relations between both temperatures.

REACTOR CONTROL SYSTEM

The nuclear power station is equipped with a relatively large system of control which ensures the normal operation of all three blocks within a range of 66% to 100% of the normal output.

A simplified line diagram of the control system of the main operational equipment of the power station is shown in Fig. 1. Three blocks operating in parallel are replaced by a single block. The control system consists of four mutually coupled control loops.

The reactor has two control loops:

1. *Neutron output control* (n) limited to a pre-set value. The regulator in this loop is characterized by the transfer of the reactor kinetics W_{kin}. Besides control rods the reactivity is influenced by the variation of the uranium temperature t_{us} through its temperature coefficient TC_u. The neutron output regulator is designed as a variable regulator ensuring control in the range of several per cent to 100%.

2. *Control of the gas outlet temperature* t_{c2}. The temperature t_{c2} is affected by the following factors: heat evolved, proportional to the neutron density n, gas temperature of the reactor inlet t_{c1} and the coolant flow-rate G. The loop of the temperature regulator W_{rt} includes, beside the transfer of the active zone

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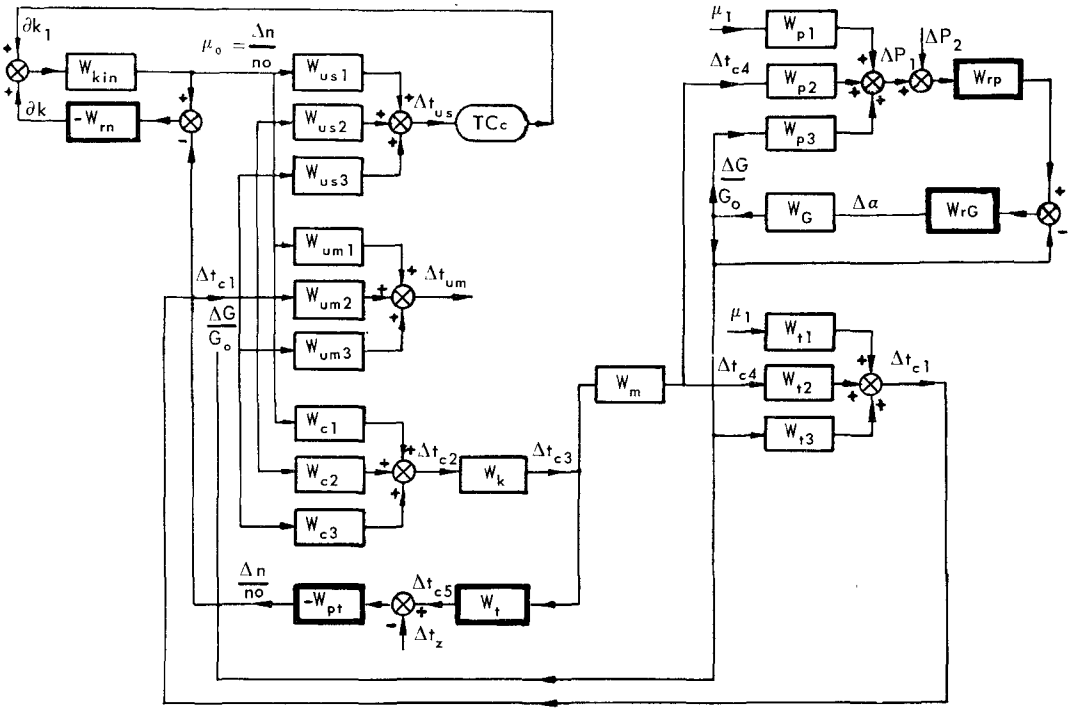


Figure 1. Block diagram of the control of the main operational equipment of the nuclear power station with gas-cooled heavy-water reactor

Heavy-outlined boxes indicate controlled transfer, other boxes indicate transfer of the controlled substance

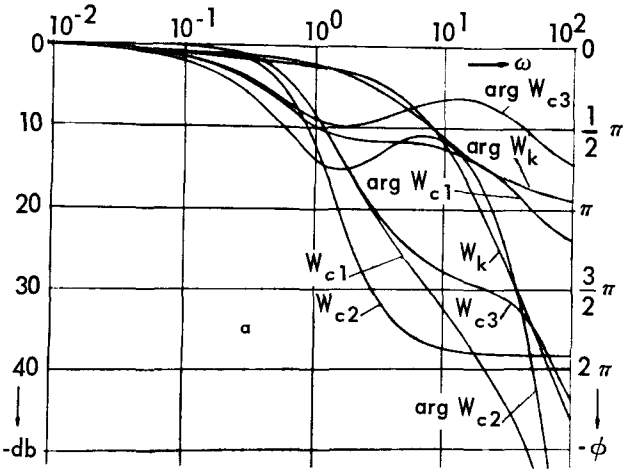


Figure 2a. Normalized amplitude-phase characteristics of the gas temperature at the reactor outlet

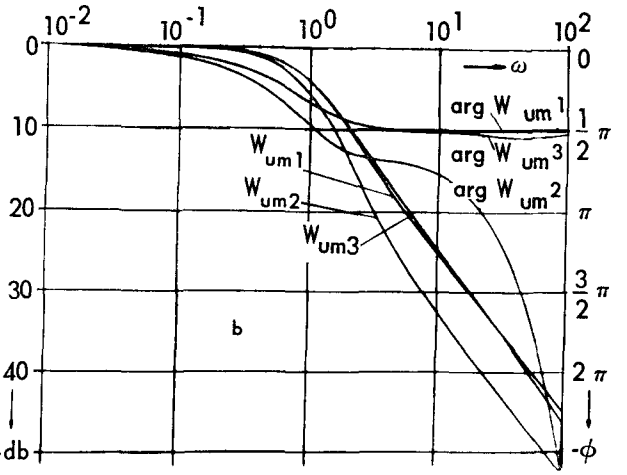
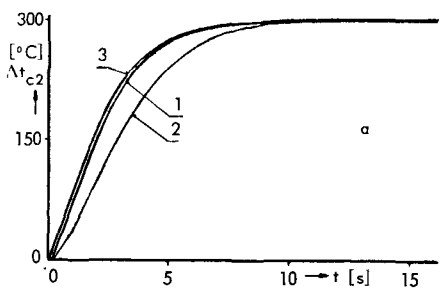


Figure 2b. Normalized amplitude-phase characteristics of the maximum uranium temperature



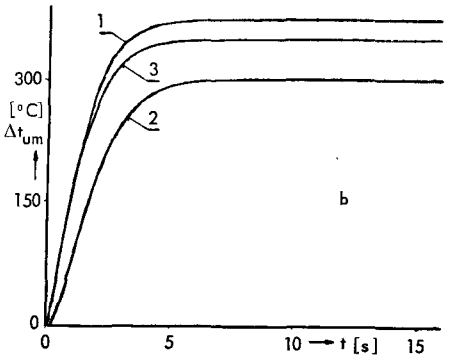


Figure 3. (*a*): Transfer characteristics of the gas temperature deviation Δt_{c_2} at the reactor outlet; (*b*): Transfer characteristics of the maximum uranium temperature deviation, Δt_{um} for the highest stressed rod

Curve 1: For disturbance $\Delta n/n_0 = 1$; Curve 2: For disturbance $\Delta t_{c_1} = \Delta T = 300\,^{\circ}C$; Curve 3 : For disturbance $\Delta G/G_0 = 1$

Table 1

Transfer of gas temperature at the reactor outlet (2)	gain coefficient	coefficients
$W_{c1} = n_o \frac{\overline{\Delta t_{c2}}}{\overline{\Delta n}} = \Delta T \frac{w_2}{k_1}\left(\pi\varphi \sin\frac{\pi\varphi}{2} - w_1 \cos\frac{\pi\varphi}{2} \right) W_{c2} + \left(\pi\varphi \sin\frac{\pi\varphi}{2} + w_1 \cos\frac{\pi\varphi}{2} \right)$	$\Delta T = 300\,^{\circ}\mathrm{C}$	$\tau_c = 0.12\mathrm{s}$
$W_{c2} = \frac{\overline{\Delta t_{c2}}}{\overline{\Delta t_{c1}}} = \exp(-w_1)$	1 /–/	$\tau_u = 0.71\mathrm{s}$
$W_{c3} = G_o \frac{\overline{\Delta t_{c2}}}{\overline{\Delta G}} = -(0.2p\tau_u + 1) W_{c1}$	$-\Delta T = 300\,^{\circ}\mathrm{C}$	$\tau_t = 0.50\mathrm{s}$
		$K_u = 3.92$
		$K_t = 0.487$
		$\pi_4 = 3.028$
Transfer of maximal uranium temperature of the highest-stressed uranium red (2)		$\tau_c = 0.09\mathrm{s}$
$W_{um1} = n_o \frac{\overline{\Delta t_{um}}}{\overline{\Delta n}} = \Delta T \frac{w_2}{k_1}\left[\frac{1}{K_u}\cos\pi\varphi\left(\xi - \frac{1}{2}\right) + \pi\varphi \sin\frac{\pi\varphi}{2} - w_1 \cos\frac{\pi\varphi}{2} W_{um2} + w_2 \frac{\pi\varphi \sin\pi\varphi\left(\xi - \frac{1}{2}\right) + w_1 \cos\pi\varphi\left(\xi - \frac{1}{2}\right)}{(w_1)^2 + (\pi\varphi)^2} \right]$	$375\,^{\circ}\mathrm{C}$	$\tau_u = 0.58\mathrm{s}$
$W_{um2} = \frac{\overline{\Delta t_{um}}}{\overline{\Delta t_{c1}}} = w_2 \exp(-\xi w_1)$	1 /–/	$\tau_t = 0\mathrm{s}$
$W_{um3} = G_o \frac{\overline{\Delta t_{um}}}{\overline{\Delta G}} = \frac{0.2\,\Delta T}{k_1 K_u}\cos\pi\varphi\left(\xi - \frac{1}{2}\right) - (0.2\tau_u p + 1) W_{um1}$	$-370\,^{\circ}\mathrm{C}$	$K_u = 2.72$
		$K_t = 0$
		$\xi = 0.742$

$$w_1 = \tau_c p + \frac{K_u \tau_u p}{\tau_u p + 1} + \frac{K_t \tau_t p}{\tau_t p + 1} \qquad w_2 = \frac{1}{\tau_u p + 1} \qquad k_1 = \frac{2 \sin\frac{\pi\varphi}{2}}{\pi\varphi} \qquad \xi = \frac{1}{\pi\varphi} \operatorname{arc\,tg} \frac{K_u}{\pi\varphi} + \frac{1}{2}$$

Transfer of outlet chamber

$$W_k = \frac{\overline{\Delta t_{c3}}}{\overline{\Delta t_{c2}}} = \frac{1}{0.0208p^2 + 0.452p + 1}$$

Transfer of communication mains reactor-steam generator

$$W_m = \frac{\overline{\Delta t_{c4}}}{\overline{\Delta t_{c3}}} = \exp(-8.1p) \cdot \exp\left(-0.65 \frac{145p}{145p + 1}\right)$$

Transfer of medium steam pressure

$$W_{p1} = \frac{\overline{\Delta p_1}}{\overline{\mu_t}} = -22.8 \frac{5.3p + 1}{70p + 1} \qquad W_{p2} = \frac{\overline{\Delta p_1}}{\overline{\Delta t_{c4}}} = 0.111 \cdot \exp(-5p) \frac{122p^2 + 2.8p + 1}{(90.5p + 1)(470p^2 + 45p + 1)}$$

$$W_{p3} = G_o \frac{\overline{\Delta p_1}}{\overline{\Delta G}} = 24.9 \frac{65p + 1}{(85p + 1)(50p + 1)}$$

NOTE: ΔT is the temperature gradient over the reactor. The transfers of the mean uranium temperature Δt_{us} are not given in the table, since their influence is negligibly small due to the low temperature coefficient of uranium reactivity, their general forms are given in [2]. The gas temperature at the outlet of the steam generators Δt_{c1} varies very little in the case of disturbances of the steam generator, the time constants are large.

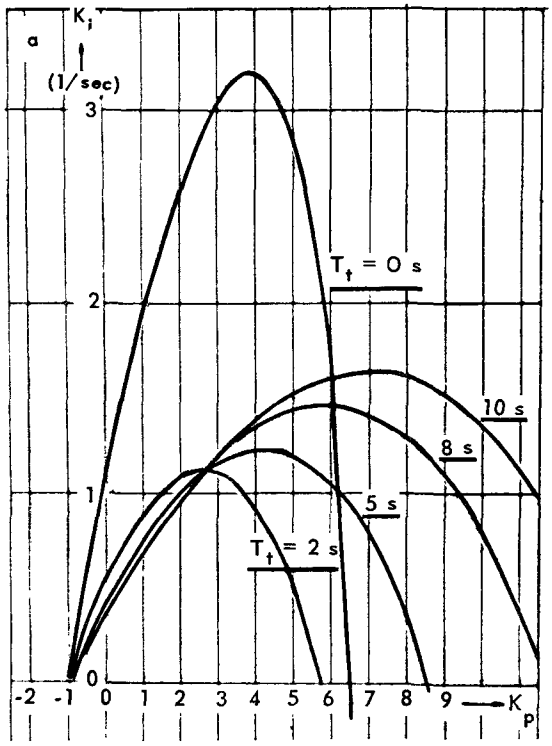


Figure 4a. Stability regions of the control circuit for the reactor outlet gas temperature, for various time constants of the thermocouple T_t (stable regions shown by hatching)

The transfer of the temperature controller is defined as $\Delta n \times \Delta T/n_0 \times \Delta t_{c3} = K_p + K_i/p$ (K_p being the proportionality constant of the temperature controller, and K_i the integration constant of the temperature controller)

W_{c1} also the transfer of the outlet chamber and the piping section up to the thermocouple W_t, and the transfer of the whole neutron density control loop. The gas temperature regulator is superior to the neutron density regulator, and its output signal requires a set value of the neutron output.

The control system for the three blocks: steam generator, blower and turbine, operating in parallel, is equipped with two control loops in each block:

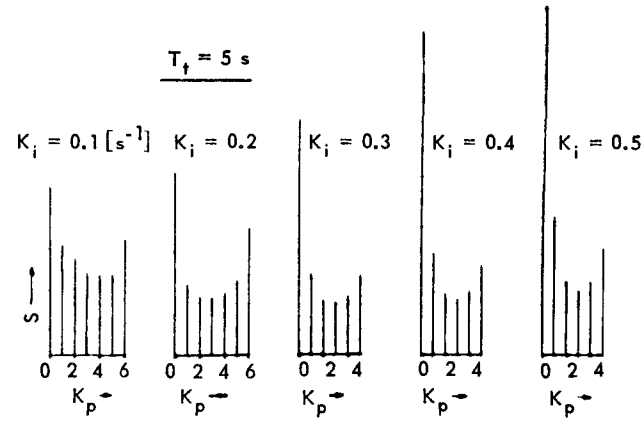


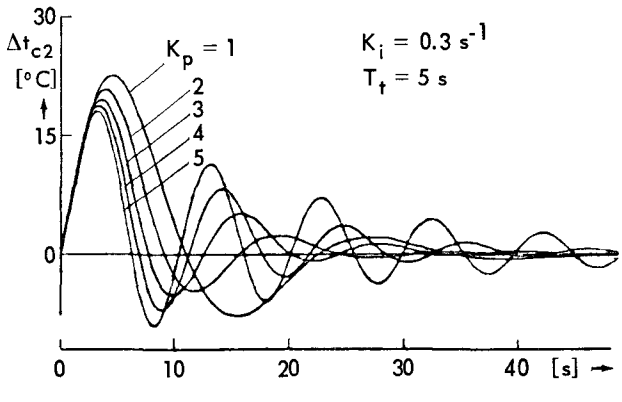
Figure 4b. Quadratic control area S for the gas temperature at the reactor outlet, for various coefficients, K_p and K_i, T_t = 5 seconds

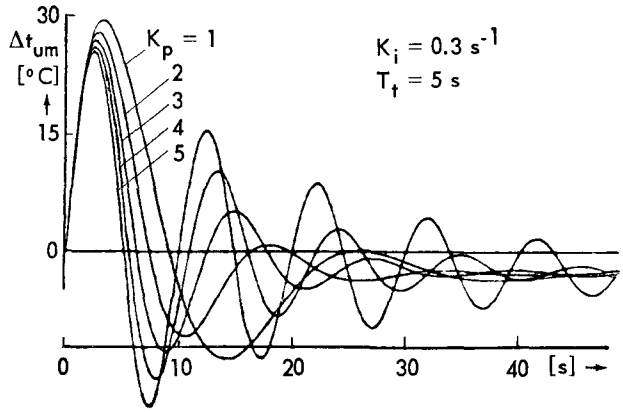
1. *Gas flow-rate regulator* W_{rG}. This acts upon the control mechanism and variations of the angle α of the guide blades of the blower vary the gas flow-rate G. Transfer time constants in the gas circuit W_G are negligibly small [3].

2. *Medium steam pressure regulator* W_{rp}. This sets the gas flow-rate simultaneously for both circulation blowers in the block. Steam pressure p_1 is influenced by the following factors: gas temperature at the steam-generator inlet t_{c4}, gas flow-rate G and the opening of the turbine valve μ_t. Since the gas temperature from the reactor outlet is kept at a constant value, no special steam temperature control is necessary in front of the turbine.

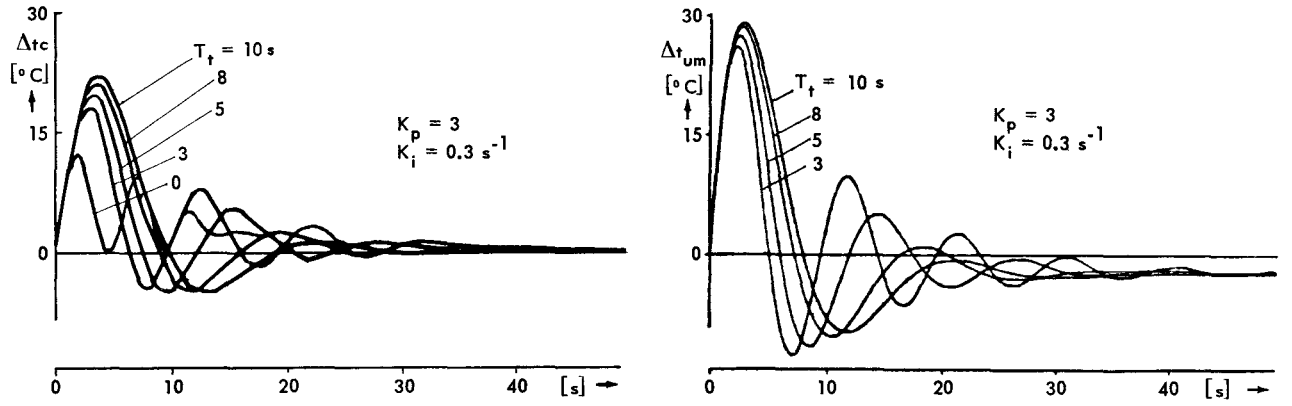
The reactor outlet gas-temperature regulator and the gas flow-rate regulator have a pulse-type output signal (+ 1, 0, −1) and they act on constant-rate servo-motors.

Transfer values of the reactor and the steam generator are given in Table 1. The temperature coefficient of uranium reactivity is relatively low, $TC_u = 3.10^{-5}$ $\delta k/°C$, therefore, in reactor control under normal conditions it has no substantial influence on the trans-





Figures 4c and 4d: Time course of the deviation of the gas temperature at the reactor outlet Δt_{c2} and of the maximum uranium temperature for disturbance $\Delta G/G_o = 0.1$ and $T_t = 5$ seconds



Figures 4e and 4f: Time courses of the deviations of the gas temperature at the reactor outlet Δt_{c_2} and maximum uranium temperatures Δt_{um} for various values of the thermocouple time constant T_t, with $K_p = 3$ and $K_i = 0.3$ second^{-1}

fer processes. The frequency response characteristic and unit function response of the gas temperature and of the maximum uranium temperature for constants in Table 1 are given in Figs. 2 and 3. The unit function responses for disturbances due to the neutron output and gas flow-rate for the gas outlet temperatures differ only very little and their time constants are 3 to 4 seconds. The cause of the rapid responses are the short time constants for the gas, $\tau_c = 0.12$ s (mean time of gas passage through the reactor), uranium $\tau_u = 0.71$ s and fuel cans $\tau_t = 0.50$ s. The gain coefficient 300 °C is very high and causes large deviations of the gas temperature with relatively low flow-rate variations.

The unit function response of the maximum uranium temperature has been determined for the rod constants which are subjected to the highest stress and their time constants are shorter and gain coefficients larger in comparison with unit function responses of the gas outlet temperature. Delays occurring in the inlet chamber are relatively low, varying around 1 s.

A parameter having an important influence on control processes in the temperature control loop is the time constant of the thermocouple T_t. Figure 4a shows stability regions calculated by means of a digital computer for transfers from Table 1 dependent up on the proportionality constant K_p and the integration constant K_i of the temperature regulator for different thermocouple time constants. When the thermocouple time constant rises from zero, the stability region first substantially decreases and then again increases, reaching a minimum of about 2 s. Figures 4b to 4f show some results of the analysis of a gas temperature control loop carried out by means of an analogue computer. The analysis has shown that an increase in the proportionality component of the temperature regulator K_p decreases the first maximum temperature oscillation and speeds up the control process. The integration component K_i has practically no influence on the temperature deviation, its increase causes first the acceleration of the control process followed by increased oscillation. A decrease of the thermocouple time constant causes a decrease of the

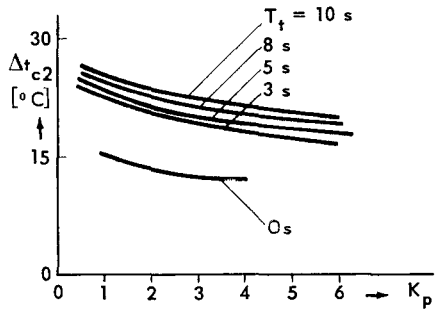


Figure 4g. Dependence of the maximum deviation of the gas outlet temperature Δt_{c_2} on the proportionality constant of the temperature controller K_p and the thermocouple time constant T_t for disturbance $\Delta G/G_o = 0.1$

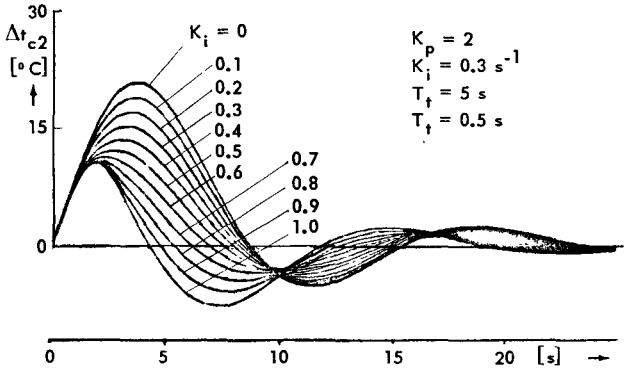


Figure 4h. Time course of the deviation of the gas temperature at the reactor outlet Δt_{c_2} for the disturbance $\Delta G/G_o = 0.1$ with the introduction of a complementary coupling from the gas flow-rate to the neutron input controller, with the transfer $G_o \times \Delta n_z/n_o \times \Delta G = K_1/(T_1 \times p + 1)$

control deviations of the gas and uranium temperatures, but with low time constants in the range of to 3 s the control process being less damped. The dependence of the first maximum gas temperature oscillation on the proportionality component K_p and the thermocouple time constant is shown in Fig. 4g. It follows from this dependence that the given gas temperature control scheme does not permit any substantial influence on the maximum deviations of gas and uranium temperatures by means of setting regulator constants when the gas flow-rate varies. The introduction of another pulse from the gas flow-rate to the neutron input regulator allows a substantial decrease of gas and uranium temperature oscillations (Fig. 4h).

Failures of the neutron output regulator and gas temperature regulator are most dangerous in operation and they generally necessitate a stoppage of the reactor due to overheating of the gas. In the nuclear power station described, however, the whole control system becomes unstable if the temperature regulators are made inoperative. Since then the temperature of the gas does not have a constant value, the medium pressure steam is influenced by a further parallel coupling from the gas flow-rate through the reactor. This coupling always decreases the gain coefficient of the medium pressure steam control loop and in the case of a large temperature gradient on the reactor its sign will change, leading to a positive feed-back in the steam pressure control loop.

A knowledge of the behaviour of the operational equipment, control and security systems of the power station in the case of failures of various types is exceptionally important both for a nuclear power station designer and operator. For the power station type under discussion, the following are taken to be the gravest causes of breakdown: (*a*) inadvertent removal of control rods; (*b*) damage to the primary pipework; (*c*) failure of the equipment power supply; (*d*) breakdown caused by the disconnection of one or both blowers; (*e*) breakdown of the turbogenerator.

The effects of these exceptional situations have been analysed in detail by means of digital and analogue computers in an attempt to obtain the maximum possible amount of information about the dynamics of transfer processes in the operational equipment in cases of a major breakdown.

We shall now discuss individual cases of breakdown and state briefly the results of the analysis:

Inadvertent removal of all control rods at a maximum rate

It is assumed that up to the moment of this accident the reactor operates at nominal output, the control rods being inserted at the point of their maximum efficiency. The safety rods will come into operation at a certain set value of Δn_A. The time interval between

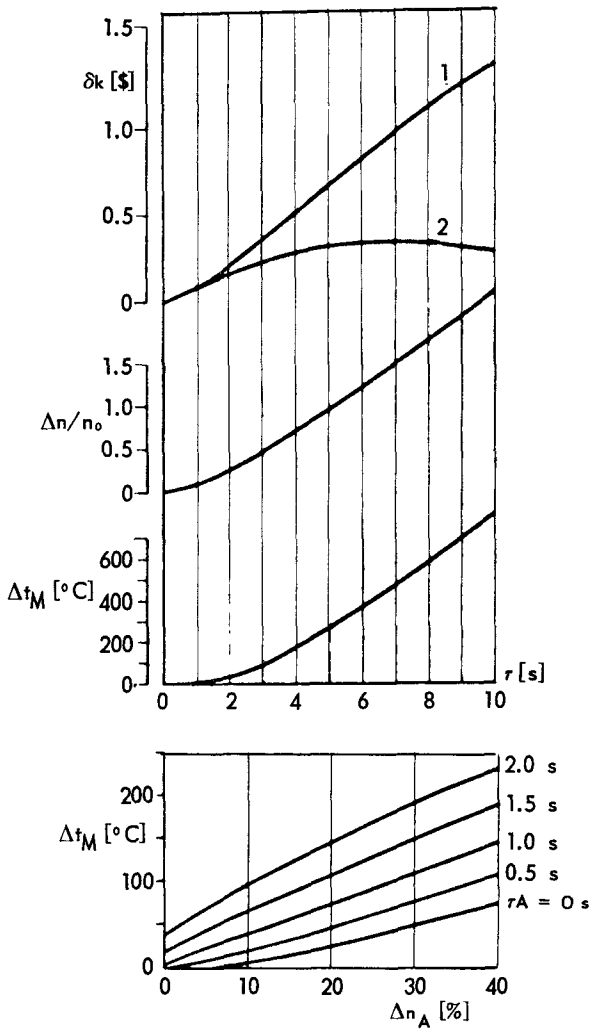


Figure 5. At top: Time course of the reactivity δk. 1: Without the influence of temperature; 2: With the influence of temperature; deviations of the reactor output Δn and deviations of the maximum temperature of the fuel Δt_M in the case of inadvertent removal of control rods; At bottom: Dependence of a transient increase of the maximum fuel temperature Δt_M on the set value of Δn_A and the delay of break-down protection τ_A

the signal and the effective operation of the safety rods τ^A is significant. Figure 5 (at top) shows the typical course of reactivity δk, reactor output and maximum temperature on the fuel element surface t_M for this case. Figure 5 (at bottom) shows the dependence of a transient increase of the maximum temperature at the fuel element surface Δt_M on Δn_A and τ_A.

The results of the analysis have served as one of the sets of data for determining the size of the pulse-type security valves of the pressure vessel.

Rupture of the main pipe-work

The analysis has shown that the reactor control system would keep the fuel surface temperature within permissible limits in spite of a relatively large amount of damage to the pipe-work. When such a breakdown occurs, however, the reactor is shut-down.

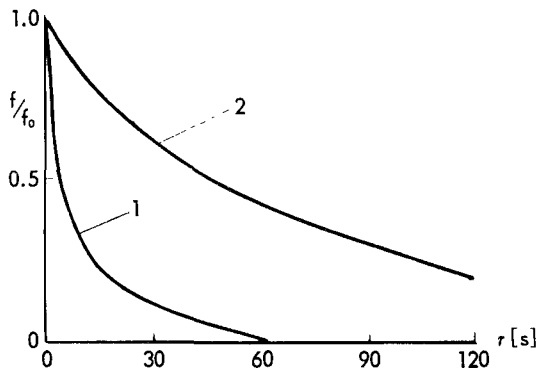


Figure 6. Run-out curve of the blowers

1: Blowers only; 2: With the use of the turbogenerators

Interruption of the reactor power supply

This will lead, among other things, to a reduction of the circulation blowers cooling capacity. In order to avoid overheating of the fuel element surface, the reactor is immediately shut-down by means of the safety rods. The remaining heat is removed for a period of 30 seconds by the blowers running under inertia. The run-down curve of the blowers is very steep and the electrical coupling to the turbogenerators is therefore retained as their inertia is utilised to maintain the blower revolutions (Fig. 6). After 30 seconds one blower is driven by power from a near-by water power plant where one machine is reserved for this purpose.

For the case of total failure of the blowers, the possibility has been studied of cooling the reactor by means of natural circulation of gas in the primary circuit as the blower is stopping. In this way it is possible to remove an amount of residual heat corresponding to about 4% of the nominal heat output of the reactor.

Stopping the reactor by means of the safety rods causes a dangerous thermal shock. Some parts of the reactor framework and the primary circuit are exposed to transient stress by sudden and large variations of temperatures and pressures. The maximum values of this stress may be reduced by two methods; either by suitably altering the structural design or by reducing the rate of temperature and pressure variations of the coolant.

The first method, i.e., by means of a thin-walled lining with an insulating gap of stationary or slowly moving gas in the outlet chamber of the reactor and in parts of the primary pipe-work [4], or by specially developed components with a special lining and modified sealing surfaces [5], etc., only partially solves the problem of thermal shock, since its effect is transferred to other parts of the primary pipe-work and to the inlet part of the steam generator. Also, the thermal shock on the fuel elements is not eliminated.

The second method may be carried out in principle either by slowing down the reduction of the thermal output of the reactor (programmed stopping by means of control rods, which however cannot be used in all cases of breakdown), or by rapidly limiting the coolant flow-rate through the reactor to a sufficient level below the permitted limit causing thermal stress. The rate of flow-rate variation is determined by the flow-rate characteristic, functional velocity and location of the limiting component in the primary circuit. From the point of view of operational security of the equipment it is important to know the transfer effects caused by the operation of the limiting component.

Much attention was given to a mathematical analysis of the problem [6]. The dynamics of the transfer processes in the primary circuit have been described by approximately one hundred equations, mostly non-linear differential, including the volume and heat capacity of the coolant and structural components, hydraulic resistances, inertia and compressibility of the coolant etc. The solution was obtained by means of a digital computer. Several technically possible means of limiting the flow-rate have been analysed, namely: (*a*) putting the corresponding hydraulic resistance of the gas main at the reactor inlet out of operation; (*b*) altering the amount of gas delivered by the blowers by means of control guide blades in the suction side, adapted especially for this use; (*c*) using a by-pass on the blowers, connected in parallel with the gas main.

The method given in (*a*) makes it possible to decrease the flow-rate to the required value. But due to the inclusion of a sufficient hydraulic resistance the hydraulic characteristic of the gas circuit is altered, and the necessary rapid shut-down of the limiting component causes a non-stationary hydraulic shock which endangers the stable operation of the blowers and possibly even the security of the pipe-work.

The means given in (*b*) may limit the flow-rate to about 40% of the nominal value within a few seconds. Calculations have shown that this value is insufficient. In order to achieve an effective limitation from the point of view of thermal stress it is necessary to apply at the same time the method given in (*c*), i.e., sufficient decrease of the flow-rate through the reactor by passing part of the gas through a by-pass. The operating time of limiting components must be of the order of 2 to 3 seconds. Information about transfer processes of the coolant pressure in various parts of the reactor, caused by a rapid limitation of the flow-rate and gas temperature variations, are very important for the designer. Some parts of the construction are stressed by non-stationary forces due to over-pressure which is larger than the corresponding static load. Figure 7 shows the typical course of the gas flow-rate G through the reactor, the temperature of the gas in the tubing connecting the reactor with the steam generator t_{c3} and

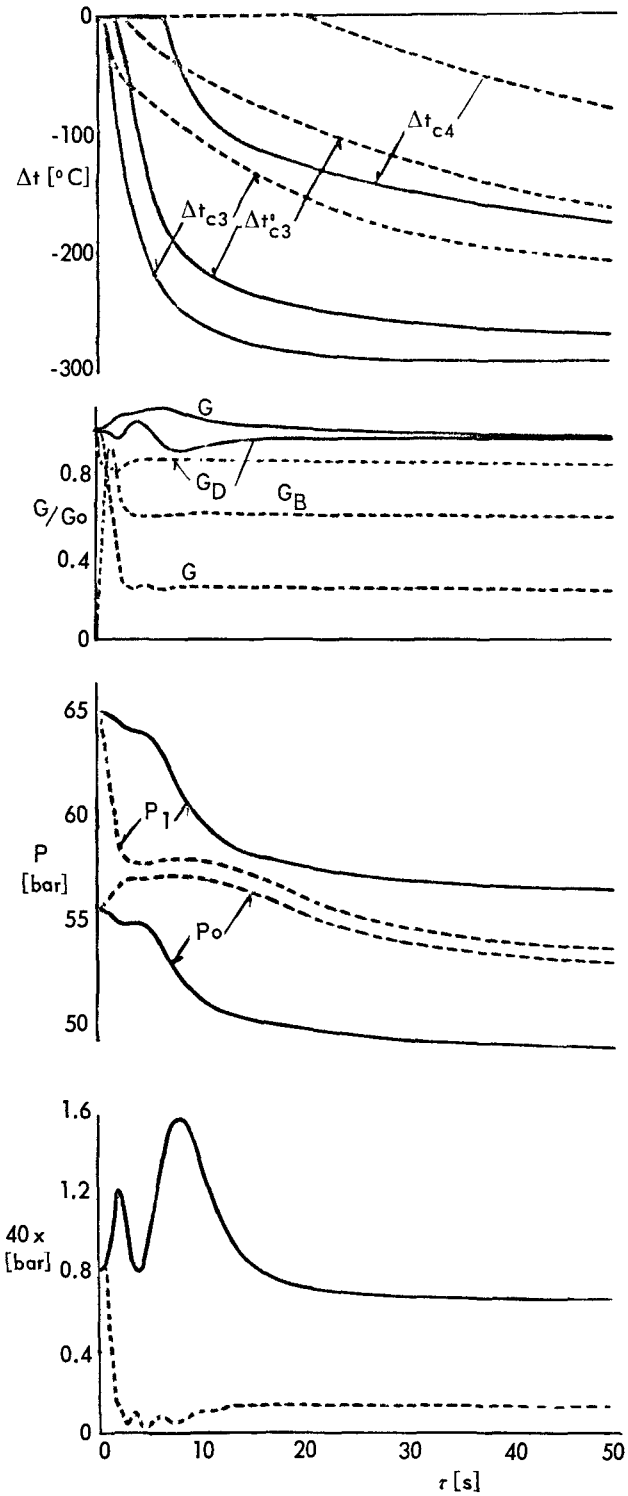


Figure 7. Time course of the deviation of the gas temperature in the mains from reactor to steam generator t_{c3}, t'_{c3} and t_{c4}, gas flow-rate through the reactor G, blower G_D, by-pass G_B, pressure of gas at the inlet p_1 and the outlet p_o of the reactor, and overpressure at the reactor inlet chamber Δp_k in the case of break-down stoppage of the reactor

Full-drawn line = without flow-rate limitation, dashed line = with limitation

t_{c4}, the gas pressure at the inlet p_1 and outlet p_o of the reactor and the over-pressure in the outlet chamber of the reactor p_k for cases without flow-rate limitation and with a limitation to about 30% of the nominal value within 2 seconds.

Breakdown caused by the disconnection of either one or two blowers

In an attempt to decrease the number of cases when the safety rods had to be used, a detailed analysis of the effects of a breakdown due to the loss of either one or two blowers was carried out. There is little probability of a breakdown of more than two blowers, thus this is not considered.

During the run-down of the blowers their operating point rapidly shifts to the pumping limit and, when this is reached, the non-return flap closes and delivery of coolant gas by this blower is stopped. Transfer processes of coolant parameters and of the maximum fuel canning temperature were analysed by calculation for the case of a breakdown of either one or two blowers. It has been found that the control system is able to cope well with a breakdown of one blower (when two blowers are damaged at the same time, the fuel canning will become unduly heated. Both cases of breakdown may, however, be solved very successfully by means of a complementary signal for a rapid decrease of the reactor output to a corresponding level.

Figure 8 shows a typical course of the reactor output n and maximum fuel canning temperature for t_M the case of a break-down of two blowers: two cases are considered.

Breakdown caused by disconnection of the turbine in the block

This does not require an immediate decrease of reactor output [7], due to the emergency set which is designed to deliver the nominal output.

When switching in the emergency set, one disconnects the normal operational control of medium pressure steam before the main turbine is disconnected and the steam pressure is controlled by opening the

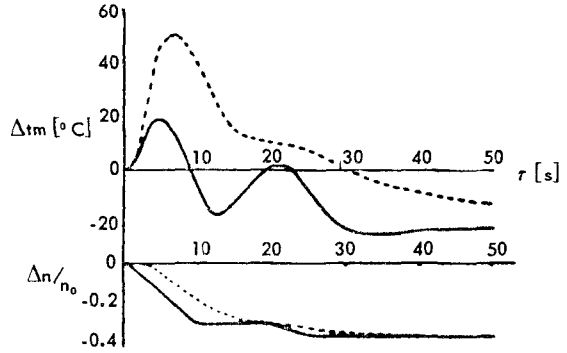


Figure 8. Time course of the deviation of maximum fuel temperature Δt_M and reactor output Δn in the case of break-down of both circulation blowers

Full-drawn line = with the control system described in operation, dashed line = with complementary signal

reducing valve of the emergency set. An integrated regulator with fixed feed-back, increasing the stability, is suitable for this purpose.

The transfer process of temperatures and pressures in the case of a breakdown caused by disconnection of the turbine depends mainly on the rate of opening the hydraulic valves which connect the emergency set to the other equipment. The whole transfer process, including the action of pulse-type safety valves, takes several tenths of a second, and it affects the reactor control only through variations of the outlet temperature of the gas from the steam generator. Gas temperature variations along the steam generator take a non-periodic course and they are negligibly small, as shown by calculations.

At other than nominal power outputs resulting from a non-linear increase in operational parameters, a change occurs in the dynamic properties of the control unit. The magnitude of the variations of the dynamic characteristics will then determine and limit the range of operation of the control system at other than nominal levels, as long as no special adaptive regulators are used. The neutron output regulator, which must give a control range of several per cent to 100% must compensate the dependence of the gain coefficient of reactor kinetics transfer at the output level.

The gas temperature control loop has some interesting properties in the other than nominal state. The gain coefficient of this circuit depends on the gas flow-rate, increasing at low flow-rate values G in the ratio of G_0/G, the time constants of uranium τ_u and of the outer channel tube τ_t being at the same time prolonged, approximately in the ratio of $(G_0/G)^{0.8}$. The result is that at a certain lower flow-rate value the operation is unstable. It the setting of the temperature regulator parameters is optimal for 100% of the flow-rate, then the limit value of flow-rate decrease is around 1/3 G_0. If correct operation of the temperature regulator is required in a broad range of gas flow-rate variations, it is necessary to use an adaptable regulator.

As shown by analysis the control loops for gas flow-rate and medium pressure steam are capable of operating satisfactorily in the required range of load variations.

The dynamic properties of a nuclear power station of this type permit rapid output adjustment in case of disturbances in the electricity distribution system. Compared to thermal power stations, the transfer process is shorter and the control range is greater.

Performance analyses have shown that the system described is capable of safeguarding satisfactory operation of the power station. At the same time, however, possibilities of further improving the system have been found (a complementary pulse from the gas flow-rate; sensitive gas temperature regulator etc.), which are being studied and analysed at present.

ACKNOWLEDGEMENT

The authors recall with pleasure their period of work in the Central Research Institute of Complex Automation in Moscow where foundations for the solution of the above problems were laid in creative co-operation with Soviet specialists. The authors also wish to thank Dr. Ing. Ševčík, Head of the Nuclear Energy Administration, for reading the manuscript and for a valuable discussion of the problems.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/525 Tchécoslovaquie

Sur quelques propriétés dynamiques d'une centrale équipée d'un réacteur à eau lourde refroidi par un gaz

par P. Stirsky et J. Hulovec

Le mémoire est consacré à la première centrale nucléaire tchécoslovaque. Dans l'introduction, les auteurs analysent brièvement les propriétés dynamiques des principaux éléments technologiques et décrivent les organes de contrôle qui stabilisent le fonctionnement du réacteur pour une puissance donnée du groupe turbo-alternateur.

Dans la partie suivante, ils analysent certains régimes transitoires de la centrale, dus à des pannes de l'équipement technique et du mécanisme de réglage.

Un arrêt accidentel du réacteur provoqué par le système de sécurité entraîne des variations impor-

tantes de température et de pressions dans le circuit primaire. Il en résulte des contraintes, supplémentaires dans certains éléments de structure du réacteur. Les auteurs examinent différentes méthodes permettant de diminuer et de limiter les chocs thermiques, les unes concernant l'agencement de l'installation et les autres consistant à réduire rapidement la circulation du fluide de refroidissement.

La défaillance d'une ou de deux soufflantes du circuit principal est une panne probable, qui entraîne une diminution considérable du débit du fluide de refroidissement dans le réacteur. Les organes de réglage du réacteur prévus pour un régime normal ne sont pas à même de remédier à ces défaillances sans que la température de la gaine dépasse les valeurs admissibles. Le mémoire indique les résultats que l'on a obtenus en prenant certaines dispositions pour limiter ce type de défaillances.

Lorsque la turbine est coupée accidentellement du réacteur, il ne s'ensuit aucune perturbation notable dans le réacteur grâce au circuit de dérivation de la vapeur conçu pour le fonctionnement à pleine puissance. Les auteurs indiquent brièvement quelques résultats concernant le réglage du circuit de dérivation dans le cas d'une défaillance de ce genre.

Il a même été tenu compte de l'influence que pourrait avoir la défaillance de l'une des 12 boucles à tubes sur l'efficacité des organes de réglage.

La principale avarie des organes de réglage consiste en une défaillance du régulateur de la température du gaz à la sortie du réacteur. Cette avarie entraîne une instabilité de tout le système de réglage. Les dispositions à prendre pour prévenir ce genre de défaillance sont analysées.

Les auteurs exposent brièvement les propriétés dynamiques de la centrale nucléaire en cas de fonctionnement à des puissances autres que la puissance nominale du réacteur. Ils terminent en évaluant le pouvoir de régularisation de ce type de centrale nucléaire dans un réseau électrique.

A/525 Чехословакия

О некоторых динамических свойствах атомной электростанции с реактором, охлаждаемым газом, и с тяжеловодным замедлителем

П. Штирский, Ян Гуловец

Настоящая работа касается первой чехословацкой атомной электростанции. Вначале кратко анализируются динамические свойства технологического оборудования станции и описывается система регулирования, обеспечивающая эксплуатацию турбин на заданном уровне мощности.

В следующей части анализируются некоторые нестационарные режимы работы станции, которые могут возникнуть вследствие аварий технологического оборудования или неисправностей в системе регулирования.

При остановке реактора аварийной защитой происходят значительные изменения температур и давлений в первичном контуре, которые вызывают дополнительные нагрузки на конструктивные элементы оборудования. Анализируются различные способы уменьшения и ограничения тепловых ударов как путем изменения конструкции обрудования, так и путем быстрого уменьшения расхода теплоносителя.

Наиболее вероятной аварией является отключение одной или двух циркуляционных газодувок, вызывающее значительное уменьшение расхода теплоносителя. Система регулирования реактора, предназначенная для обеспечения нормальной его работы, не может привести в соответствие это изменение с допустимыми отклонениями температуры покрытия прутковых тепловыделяющих элементов. Приводятся некоторые результаты исследования мероприятий, направленных на предотвращение аварии такого типа.

Аварийная остановка турбины при наличии РОУ, сконструированной на полный расход пара, не вызывает значительных нарушений режима работы реактора. Кратко приводятся некоторые результаты регулирования РОУ при такой аварии.

Исследовалось влияние разрыва одного из двенадцати газопроводов на работу системы регулирования.

Наиболее важным дефектом в системе регулирования является неисправность регулятора температуры газа на выходе из реактора. Такая неисправность вызывает неустойчивость всей регулируемой системы. Анализируются возможности для устранения такой аварии.

Приводятся динамические свойства станции при неноминальных режимах работы. В заключение оцениваются динамические свойства станции указанного типа в энергетической системе.

A/525 Checoslovaquia

Estudio de ciertas propiedades dinámicas de una central nuclear de reactor moderado por agua pesada y refrigerado por gas

por P. Štirský y J. Hůlovec

Este trabajo trata de la primera central nuclear checoslovaca. Se analizan brevemente las propiedades dinamicas del equipo tecnológico principal y se describe en la introducción el sistema de control que asegura la operación a un nivel de potencia dado del turbogenerador.

A continuación, los autores de la memoria analizan algunos transitorios en la central nuclear, ocasionados por fallos del equipo tecnológico y del sistema de regulación.

En ciertos casos de parada accidental del reactor, se producen grandes variaciones de temperatura y presión en el circuito primario. Este fenómeno da lugar a tensiones desfavorables en algunas partes de la estructura del reactor. Se analizan distintos procedimientos para reducir y restringir los choques térmicos, bien mediante un diseño adecuado, o por disminución rápida del flujo de refrigerante.

Desde el punto de vista de la operación, es probable el fallo de uno o de dos soplantes principales de circulación, lo que da lugar a una disminución considerable del flujo de refrigeración del reactor. El sistema de regulación, especificado para operación normal, es incapaz de contener estos fallos sin que la temperatura exceda del límite permisible para las vainas de los elementos combustibles. Se dan aquí algunos resultados de las medidas tomadas para limitar este tipo de fallo.

La parada accidental de la turbina no supone gran perturbación para el reactor, lo que se debe a un conducto de derivación del vapor diseñado para plena potencia. Se mencionan brevemente los resultados obtenidos con este sistema en la regulación en el caso de un fallo de este tipo.

Se tomó incluso en cuenta la influencia del fallo de uno de los doce tubos del circuito sobre el funcionamiento del sistema de regulación.

La avería principal del sistema de regulación es la del regulador de la temperatura del gas de salida, lo cual da lugar a la pérdida de la correspondiente señal de regulación. Este tipo de fallo provoca inestabilidad en todo el sistema de control. Se analizan diversos métodos para suprimir esta anomalía.

Se explican brevemente las propiedades dinámicas de la central nuclear durante el funcionamiento del reactor a niveles de potencia distintos del nominal. Por último, se agrupan las propiedades más destacadas de regulación de este tipo de central nuclear en el sistema de energía eléctrica.

The development of burst cartridge detection systems with particular reference to Sizewell nuclear power station

by B. Holmes and A. D. Boothroyd *

The purpose of burst cartridge detection (BCD) systems is to enable the coolant activity level to be kept acceptably low. In order to achieve this, it must be possible to locate faulty elements, as well as to detect their existence, so that they may be removed from the reactor before the activity becomes excessive. Thus the background activity level be must kept at a sufficiently low level to allow identification of the defective channels.

When the reactors of Calder Hall and the first few commercial nuclear power stations in the UK were designed, no operating experience existed to indicate the types of fuel element failure which would occur. The following fundamental requirements for BCD formed the system specification:

(*a*) That the bulk gas activity be known at all times;

(*b*) That it should be possible to sample each channel individually; and

(*c*) That a routine scan of the reactor be carried out at such sensitivity and frequency as to permit removal of the faulty elements before an unacceptable activity level is reached.

In each design, the first requirement was met by a bulk gas monitor and the second by a manual selection system. A third system was designed to scan continually groups of reactor channels, the number in each group and the period of a complete scan being determined somewhat arbitrarily.

Operation of the reactors at Calder Hall and Chapelcross [1, 2] subsequently showed that fast bursts, giving virtually instantaneous release of a significant quantity of fission products and rapid bursts, growing to unacceptable size within a few hours, need particular consideration. For fast bursts, quick detection and location are important and the discovery of this type of burst indicated a need to increase the frequency of group scanning at the expense of sensitivity. In subsequent designs, therefore, larger number of channels have been grouped together and these groups scanned more frequently.

An appreciable number of slow or cavitation failures have also occurred at Calder Hall and Chapelcross. Development of more ductile, finer-grain canning materials reduces the susceptibility of current fuel to these failures. However, there is an economic incentive to operate the fuel up to its metallurgical limits and a number of small leaks must be considered probable. Such failures would lead to a gradual rise in background; and detection of individual leaks would not always be possible using a scanning system grouping a lot of channels together. In order to operate the reactor with a number of such leaks, it is necessary to be able to locate each leak ant to check its size and rate of growth regularly. This has been made possible on the more recent reactors by the provision of equipment to perform a fully automatic scan of individual channels.

Since Calder Hall and Chapelcross experience became available, all BCD systems designed for uranium/Magnox reactors in the UK have employed the combination of a fast scan at low sensitivity with a single channel scan of long cycle time. Some details of the systems for recent reactors are given in Table 1.

SYSTEM SPECIFICATION FOR SIZEWELL

The performance of a BCD system needs to be judged in conjunction with the time required to discharge a defective channel. The refuelling system at Sizewell differs from earlier stations in that charge chutes are left permanently installed in the reactor. The arrangement is illustrated in Fig. 1. Guide tubes above the core bring the channel outlets to a circular configuration, suitable for a simple charge chute. Control rod mechanisms and other equipment installed from pile cap are fitted into the same standpipes as the charge chutes but are not required to be removed when refuelling. This system allows channels to be refuelled using only one machine and eliminates the need to insert first a charge chute, by either a separate machine or a complex combined machine. With this equipment it is possible to discharge a channel of fuel within 30 minutes if the fuelling machine is immediately available or within a maximum of 110 minutes if the machine is already in use.

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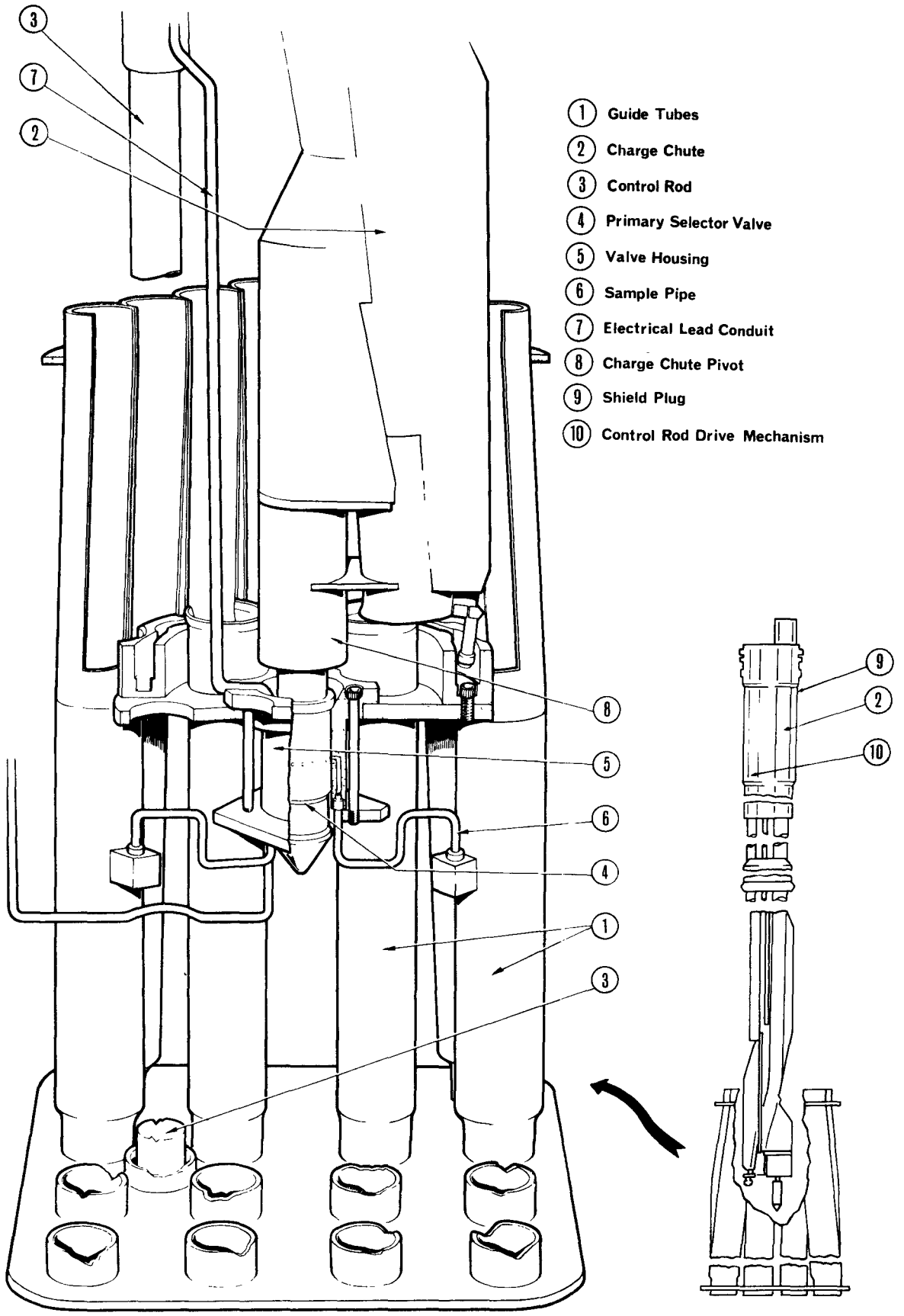


Figure 1. Sizewell fuelling equipment

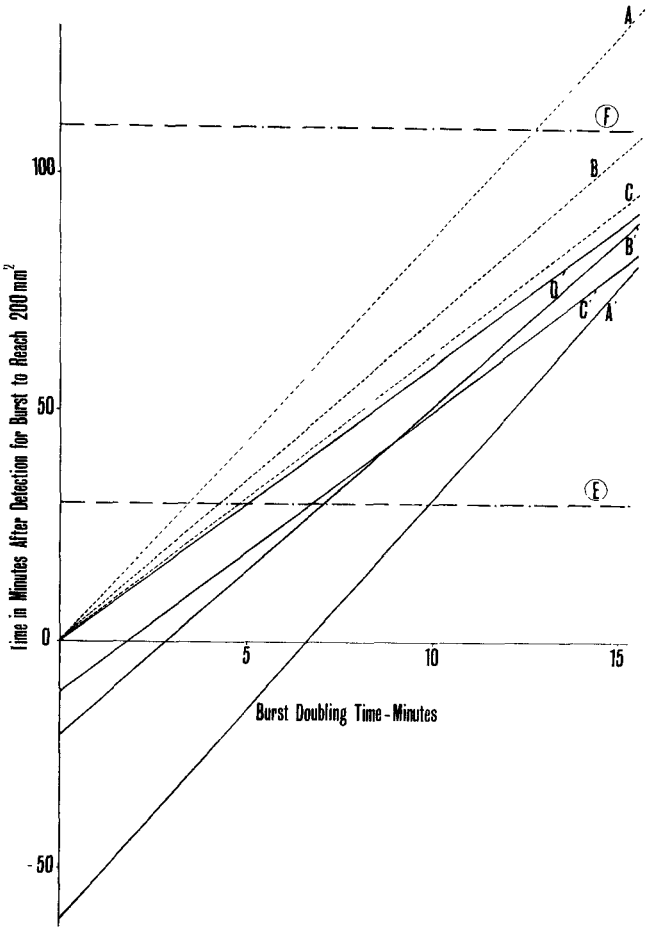


Figure 2. Performance of systems compared

The specification of the Sizewell BCD system followed a study of a range of possible group sizes and scan times. The signal doubling time is taken as the characteristic of an element failure. The time interval between the moment of detection and the time at which the burst reaches a level requiring reactor shut-down indicates system performance. An arbitrary shut-down level equivalent to a 200 mm² uranium foil exposed at the centre of the centre channel is used in the comparison of various systems given in Fig. 2.

Curves A and A′ relate to a system in which single channels are scanned at intervals of 1 h. Curve A represents the best performance, assuming that the channel is scanned as soon as the burst reaches a detectable size. Curve A′ is the worst performance, assuming that the burst reaches detectable size immediately after the channel has been scanned and that a full scanning period elapses before detection actually occurs. Curves B and B′ are the corresponding characteristics for a system scanning groups of 8 channels at intervals of 20 min. Curves C and C′ relate to groups of 16 channels scanned at intervals of 10 min. Curve D refers to the performance of a system continuously monitoring groups of 32 channels. The horizontal lines marked E and F indicate respectively the shortest and longest times required for discharge of the channel.

Table 1. Main parameters of recent BCD systems

	Dungeness	Sizewell	Oldbury	Wylfa
Reactor No. 1 on load	1964	1965	1966	1968
Channels in reactor	3 932	3 784	3 308	6 156
Channels per group sample	32	32	8	32
Period of group scan: min	6.7	Continuous	11	Continuous
Cycle time per sample: s	25	60	25	60
Single channel scan period: h	3.7	4.25	4.0	7.1
Total precipitator units: per reactor	20	18[a]	27	29[a]
Group sample flow rate at precipitator: lb/s	0.115	0.037	0.115	0.031
Single sample flow rate at precipitator: lb/s	0.042	0.017	0.042	0.026
Total circuit mass flow: lb/s	1.6	5.2	2.4	7.3
Circuit pressure drop: psi	40–45	40	42–47	55
Transit time: channel to precipitator: s	6	5.9	11	6.7
Sensitivity (MDA)[b]:				
Group mm²	10.0	3.5	2.5	3.8
Single mm²	0.57	0.11	0.57	0.13

a Eight chambers per precipitator unit.

b Minimum detectable area as determined by an alarm set 25% above background for a scanned system and 15% above background for a continuous system.

For each of the scanning systems the alarm setting is assumed to be 25% above background. This figure is based on an operationally acceptable number of spurious alarms due to random variations in the integrated count. Since the reactor operators require confirmation that an alarm is genuine before initiating action to defuel a channel or to shut down the reactor, a second count must be obtained from the group for which an alarm has been raised. Thus, either a full scanning period must elapse before the alarm is confirmed, or the scan must be interrupted for a second count. This problem does not arise with a continuous monitor for which a confirmatory signal is automatically given by the next count. An increased number of spurious alarms are therefore acceptable. Furthermore, the random variations in signal are reduced since the sample flow is not interrupted. These factors permit an alarm setting at about 15% above background.

Assuming the fuelling machine to be immediately available, the continuous system ensures that bursts having a doubling time of five minutes or more can be removed without reactor shut-down. With a scanning system this ability depends on the time in the scanning period at which the burst reaches detectable level. For a doubling time of about fifteen minutes, the pessimistic curves A′, B′ and C′ and curve D indicate

similar performance. For even longer doubling times, the smaller group, longer period scans are shown to provide greater warning, but this is relatively unimportant at Sizewell as any of the systems would give adequate warning for removal of the defective fuel even with the machine occupied when detection occurs. More important is the performance in respect of bursts with shorter doubling times. Figure 2 shows that, for decreasing doubling times, curves A, B, C and D converge and sensitivity becomes relatively unimportant in comparison with the speed of detection.

The curves indicate the time available from detection to the time at which the channel must be discharged. In practice, the operator needs confirmation of an alarm before taking action to avoid inconvenience arising from spurious alarms. This takes only one minute on a continuous system, but generally takes a further cycle of a scanning system. Hence, in practice, the continuous system gives more warning than is evident from the curves.

It is evident that the risk of circuit contamination is significantly reduced by the adoption of a continuous system offering extremely rapid detection of a fast burst. Further, by arranging to alarm on the highest signal it is possible to locate the affected group without the confusion which can occur with a scanning system where the alarm may be raised by recirculated fission products drawn from a group of healthy channels. A continuous group monitor, therefore, was chosen for Sizewell. Continual operation of selector valves was eliminated and hence system reliability improved.

To monitor the channels in groups of 32, a large number of precipitation chambers are required. This would not have been economically feasible but for the multi-cell precipitation units developed for the Windscale AGR [3]. The refuelling equipment enables removable selector valves to be installed adjacent to the sampling points. These manifold the group samples and select for single channel sampling. The use of long guide tubes rather than shallow charge pans also made it possible to locate a single offtake on each tube, sufficiently high above the fuel to ensure mixing of the gas flow adequate to obtain a representative sample, even with the low sample flow used for the group monitor.

The sample gas flow block diagram is shown in Fig. 3. In the group system, samples from the channels are grouped in the primary selector valves and, after being cooled, are passed through the precipitators. Samples from individual channels can be selected by the primary and secondary valves, either for single channel scanning or for continuous observation. The duct monitoring system provides an over-all check on every channel when a portion of the group system is inoperative for any reason.

A further important factor arising from Calder Hall experience is the ability to detect and remove potential fast bursts before they contaminate the reactor. Detailed examination of fuel elements, after a fast burst, has shown that they originate as welding defects on fuel assembly or from mechanical damage during fuelling. A small leak path exists when the fuel is new. Fission products cannot diffuse out of the can as there is a constant inward flow of CO_2 due to the corrosion reaction with the uranium. Experiments have shown that these incipient fast bursts can be detected by inducing gas flow out of the elements by rapid reductions in channel pressure or fuel temperature. A continuous system has substantial advantage in this respect in that all the channels in a reactor can be observed simultaneously during changes of pressure or mass flow. In the event of the presence of a potential fast burst being suspected in a group of channels, it is a simple operation to use the fuelling machine to lower a flow restrictor into each channel in turn, to create a change of mass flow while the channel is monitored by the lock-on facility.

DESCRIPTION OF THE SIZEWELL SYSTEMS

Group monitor system

In the group monitor system, samples of gas are taken from the 16 channels associated with each fuelling standpipe. Each set of samples is combined and filtered in the primary selector valve. The combined samples are drawn through small bore stainless steel pipes to the group sample valve panel in the BCD room. In this panel, the pipes pass through isolating valves, join together in pairs and then pass through coolers. The samples from the groups of 32 channels then pass through the precipitation units. From the precipitators all gas is returned to the main reactor circuit by a reciprocating compressor.

Gas flows continuously through the system and the precipitator wires are indexed every minute. The count from each photomultiplier is compared with high and low count levels. The count values are not recorded but if any is abnormal, an alarm is raised in the main control room and the group identity indicated. A low count alarm indicates an equipment fault; a high alarm suggests a burst.

On receipt of a high count alarm, the reactor engineer initiates the burst location sequence using the single channel scan and transfers the appropriate integrator from the alarm display to a recorder. The next count recorded either confirms the alarm or suggests that it was spurious. Further points are plotted at minute intervals indicating the rate of development of the burst. Alarms are triggered during the counting period as soon as the integrated count reaches the alarm setting. If any other group reaches alarm level later in the same period, its alarm is inhibited. In the event of a fast burst, this feature ensures that the

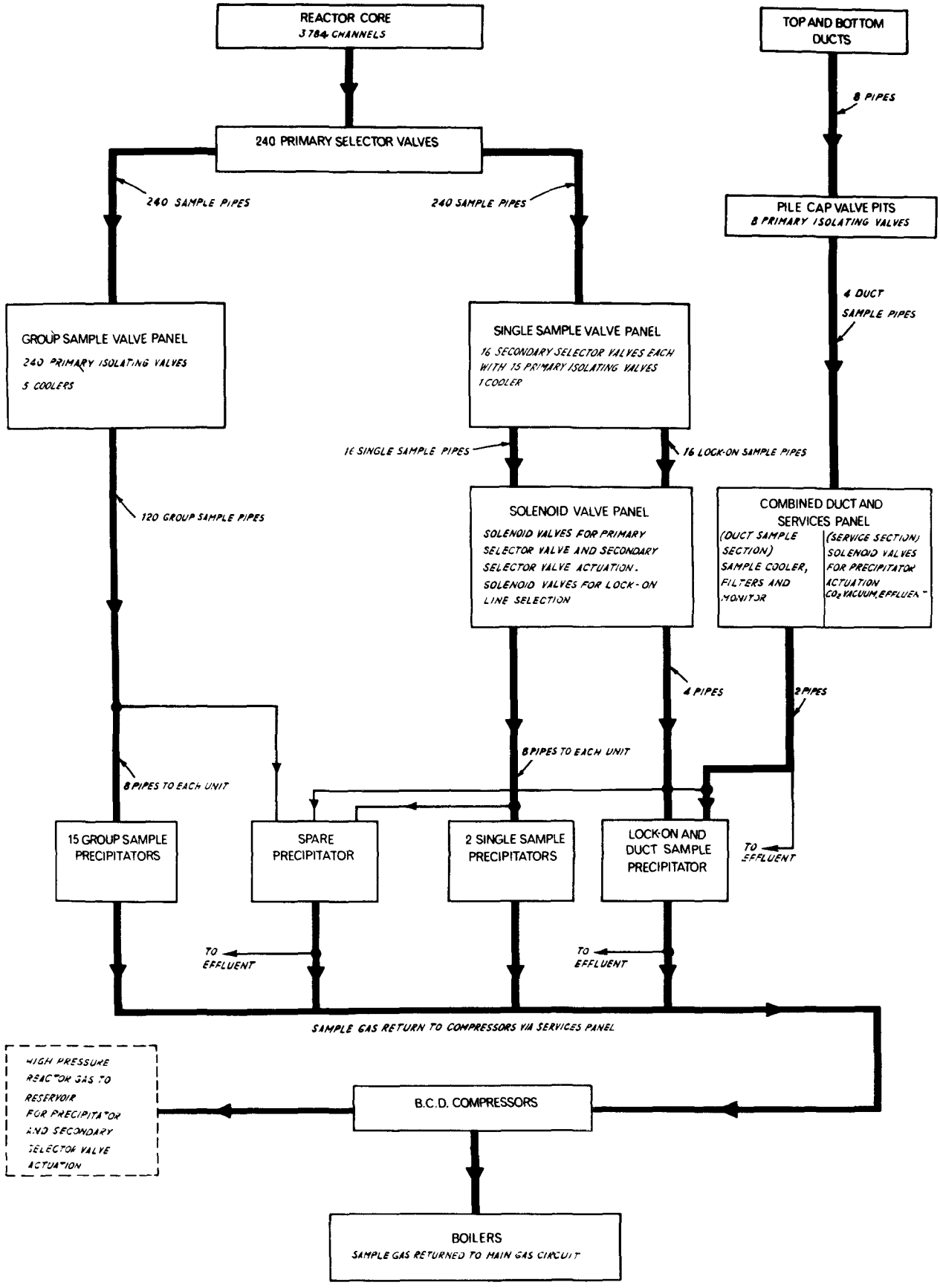


Figure 3. Sizewell BCD system block flow diagram

defective group is distinguished from groups which reach alarm level due to recirculated fission products. This feature is not feasible in a scanned system in which there is a very real chance that the first alarm would be raised by the recirculated products, thus making subsequent location difficult. When an integrator has been transferred to the recorder, the alarm display is free to indicate a further alarm. All groups having counts above alarm level can be indicated when requested by the operator.

From the above, it is seen that the first high count alarm indicates only that there is a signal which can be discriminated from the background. A second, higher alarm is provided to indicate that action is required.

Single sample system

The primary selector valves are designed to permit single filtered samples to be taken out of each group of 16 channels. A small bore pipe is taken from each selector valve to carry these samples to the BCD room. These pipes are connected into secondary selector valves, the bodies of which contain the primary isolating valves. There are 16 secondary valves, each associated with 15 primary valves. From these valves, the samples are routed through coolers to the single sample precipitators.

The sequence of operations for a single channel scan is automatically controlled and is designed so that samples from 16 channels at a time flow through the precipitators. The selector valves are actuated so that all channels are sampled in turn, a complete scan taking 4¼ hrs. It is expected that such a scan will be performed once each day although continuous operation is possible. Since the integrated counts have no immediate bearing on reactor operation, they are recorded on punched cards for subsequent analysis by computer.

For burst location following a group alarm, this system is used to scan the affected channels. When a location sequence is initiated, the secondary selector valves are indexed to the appropriate position and a scan performed using only the primary selector valves. The counts are recorded in the control room annexe, where the scan is controlled, and are repeated on the control desk for the information of the reactor engineer. When the high count from the faulty channel is indicated, the channel and standpipe are identified simultaneously but the scan of all channels is completed to ensure that only one channel has a high count.

The primary selector valves ares actuated using clean CO_2, by applying a pressure pulse to the single sample outlet pipe, thereby lifting and turning a selection rotor. This avoids any requirements for separate actuation connections to the valves. Indication of rotation is given by two pairs of electrical contacts: one indicates each movement of the rotor and the other its return to the parked position at the end of each scan. A similar actuation system is used for the secondary valves, but separate actuation lines are used and clean CO_2 is not required. The actuation pressure is obtained from the compressor outlet.

Lock-on facility

Four lock-on facilities are provided so that individual channels can be continuously monitored. They are integral with the single channel scan in that the primary and secondary selector valves are used. The secondary valves each contain a second rotor, indexed independently of the single sample rotor. In order to increase flexibility in the selection of channels for observation, it is possible to connect any secondary selector valve to either of two precipitation chambers, the choice being made by operating one of a number of solenoid valves.

The lock-on lines are set up from the main control room, the control sequence being entirely automatic. The control system selects and operates one of the solenoid valves opening a route from the secondary valve to the precipitator. The secondary and then the primary selector valves are actuated to complete the connection of the required channel. The precipitator is indexed every minute and the integrated count recorded in the control room annexe. The count on any one of the four recorders can be repeated to the main control desk and to the fuelling machine control desk, so that the reactor engineer and the fuelling machine operator can monitor a channel being refuelled.

Bulk gas monitor

Two samples of gas are taken from each of the main gas circuits, one being from the outlet duct immediately downstream of the main isolating valve and one from the inlet duct downstream of the main circulators. Sample pipes from all the ducts are routed to the BCD room where the outlet duct, and likewise the inlet duct, samples are manifolded. The samples are cooled and filtered before passing to two precipitation chambers. The counts are recorded in the control room annexe and alarmed in the main control room. They may be repeated on a recorder on the control desk, if required. This will normally be done should a portion of the group monitor be inoperative. The gas circuit is normally completed by connection to the inlet of the compressor, but, to enable the system to operate when the compressors are not available, flow can be induced by allowing the used sample gas to blow off to atmosphere via the effluent system.

The filters in this system are monitored by Geiger tubes to detect the presence of active particles in the gas circuit during reactor shut-down.

SYSTEM DESIGN AND PERFORMANCE

The group monitoring system is regarded as the most important system and emphasis has been placed on the requirements of this system when determining the details of the equipment. In optimising the system, a compromise has been achieved between system sensitivity, compressor power and cooler rating.

The length of sample pipe has been kept to a minimum by careful attention to layout both inside and outside the reactor vessel and to the location of the BCD room in relation to the reactor. Although the lengths of individual pipes differ, the variation in mass flow due to this is kept within 10% of the nominal value. Having determined the lengths of the sample pipes, the sample mass flow and the pipe size were obtained.

The majority of the gas pumped by the compressor is from the group system. Hence this system dictates the pressure difference between the reactor and the precipitator outlet header. The sample flows for the other systems are then fixed by the pressure head available and the pipe dimensions.

The parameters chosen and the resulting sensitivities are given in Table 1. The figures quoted are for a clean reactor with a background attributable largely to the contamination on the external surfaces of the fuel elements. Allowance is made for ambient gamma activity and photomultiplier noise. There will also be a stray background count attributable to shine on the phosphor from active gas in the precipitation chamber and to adsorption of active gas on the wire. The magnitude of the count from this source cannot be accurately predicted and is not included in the figures quoted.

ENGINEERING PROBLEMS IN THE SIZEWELL EQUIPMENT

The pipework, isolating and solenoid valves, coolers and compressors conform to what has become standard nuclear power station practice. The precipitators are similar to the units developed for the Windscale AGR. Only the primary and secondary selector valves were specifically developed for the Sizewell reactors. Each reactor requires 240 primary and 16 secondary valves and, as the former are situated within the reactor vessel, reliability is of great importance.

Primary selector valves

The primary selector valve is illustrated in Fig. 4. Positioned at the lower end of the charge chute (Fig. 1), it is located in a housing mounted in the guide tube assembly. The top of the valve forms the lower pivot about which the charge chute rotates. The valve contains a rotor with a shaped groove which engages with an indexing pin attached to the body. High pressure is applied to, and then released from, the underside of the rotor, causing the rotor to turn as it rises and falls. The rotor contains sintered stainless steel filters to filter both group and single samples before they leave the valve.

Two pairs of contacts are operated by the rotor at the top of its stroke, one every time the valve is indexed and the other only when the valve is indexed to its parked position. They are monitored by the control equipment to check for correct operation of the valves.

At one position on the circumference of the rotor there is a port which can be lined up with any of the 16 inlet ports in the body or with a blank position. The remainder of the circumference comprises an annulus which encompasses all other inlet ports thus providing the manifold for the group sample. The group sample flows through the upper filter and leaves the valve by the group sample outlet port. The single sample flows through the lower filter and out via the single sample outlet port. To minimise gas flow during actuation, a disc type non-return valve is incorporated in the rotor below the single sample filter.

The valves are required to operate at a temperature of about 400 °C in a CO_2 atmosphere and all rubbing surfaces are therefore nitrided. Development of the valve has included wear and leakage tests and no effect of continued operation has been found other than wear of the indexing pin. Tests have been carried out at operating conditions and uninterrupted operation has been consistently achieved for numbers of operations several times the total required for one scan per day throughout the life of the station. Pin wear is not excessive until about five times this number of operations. The valves are designed so that the filters and drive pin can be replaced using manipulators in a remote handling facility.

Secondary selector valves

The secondary selector valve is illustrated in Fig. 5. The single sample rotor provides sequential selection of primary selector valves during a single channel scan, while the lock-on rotor provides access to the primary valves for monitoring a particular channel. Both rotors are housed in a common pressure containing body. They are similar to the primary valve rotors and are entirely independent in their operation. There are, however, no filters and no manifold annulus. As there are no outlet connections to the underside of the rotors, separate actuation pipes are provided.

The operating temperature of these valves is in the range 250–300 °C. Life tests have shown similar behaviour to that of the primary valves and no measurable wear has been found on the rotors despite their horizontal disposition. Pin wear is, if anything, rather greater than in the primary valve but operation of the

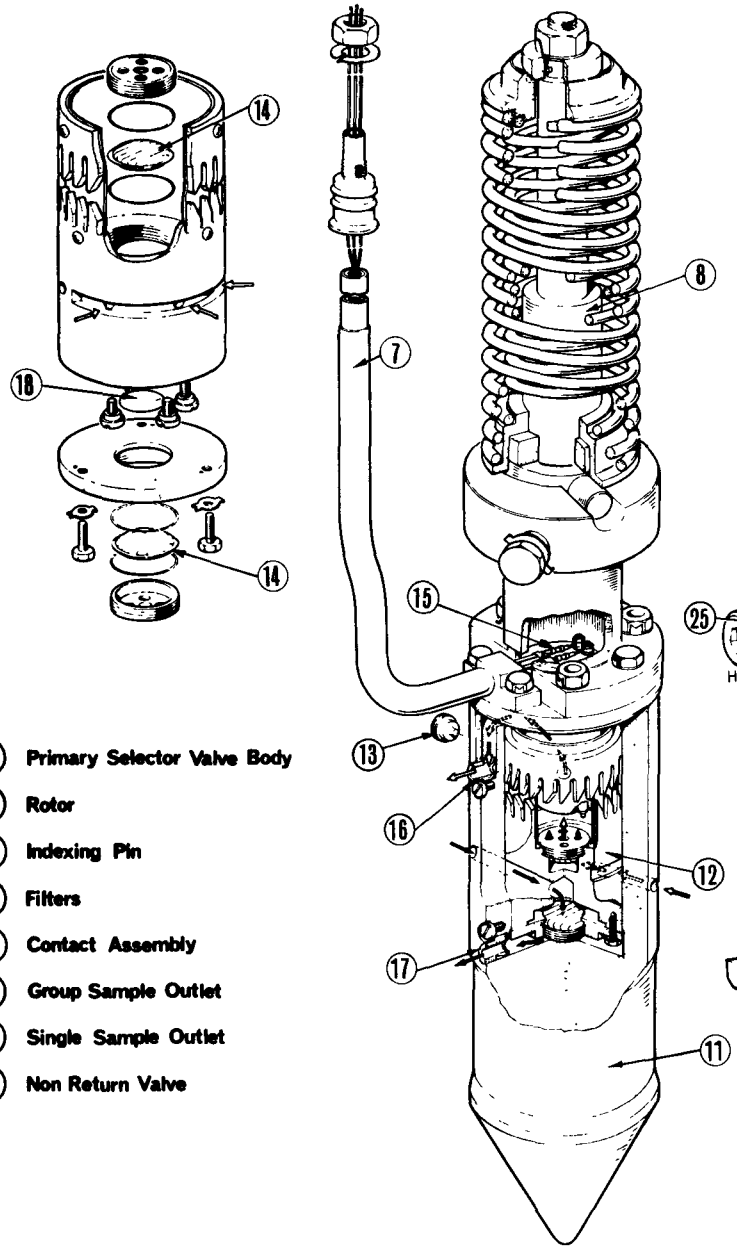


Figure 4. Primary selector valve

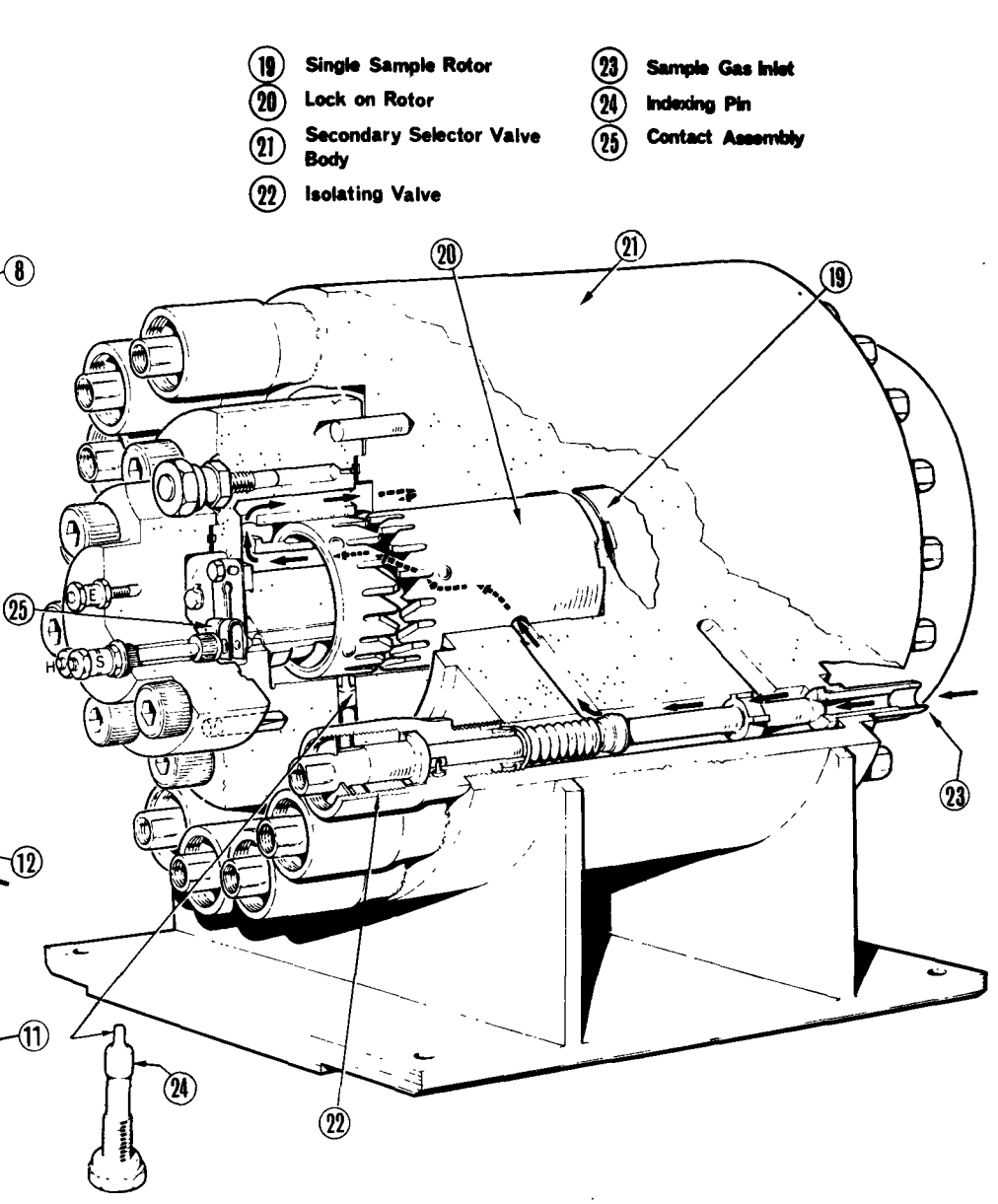


Figure 5. Secondary selector valve

secondary valve has been found to be less susceptible to this factor.

FURTHER DEVELOPMENTS

A logical development of the Sizewell group system is the replacement of the snatched collector type of precipitator by one using a continuously moving collector. Not only would this almost entirely eliminate mechanical complexity, but, by suitable design of the precipitation chamber and the operating characteristic of the ratemeter, varying sensitivity can be obtained for bursts of differing doubling times. Fast bursts would be indicated almost instantaneously at low sensitivity, but increased sensitivity would be achieved for detecting failures involving less urgency.

An even more significant development arises from the inclusion of a thermocouple in the primary selector valve. It is then possible, either in conjunction with a BCD single channel scan or separately, to measure the temperature of the sample gas and hence to determine the channel gas outlet temperature for every channel in the reactor. By appropriate programming it is possible to obtain a fast representative survey of, say 1: 8 or 1: 16 channels, or to carry out a slower, more detailed survey of the entire reactor. These thermocouples do not suffer the disadvantage of the more normal installation for gas temperature measurement since each can be replaced if defective at any time during the life of the reactor.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/142 Royaume-Uni

Mise au point de systèmes de détection de rupture de gaine, notamment pour la centrale nucléaire de Sizewell

par B. Holmes et A. D. Boothroyd

L'exploitation des réacteurs de Calder Hall et Chapelcross a montré que les systèmes de détection de rupture de gaine doivent être conçus pour détecter et localiser les ruptures rapides et très rapides et les défaillances par cavitation. La nécessité de détection urgente des ruptures rapides donne la priorité à la fréquence du balayage aux dépens de la sensibilité. D'autre part la nécessité de localiser les canaux contenant des défauts par cavitation, qui contribuent à une lente élévation des bruits parasites, exige que l'on puisse balayer le réacteur avec une sensibilité élevée.

Pour les réacteurs récents un système de balayage double a été adopté. Dans ce système un balayage rapide par groupes, allant jusqu'à 32 canaux, fournit une protection constante normale, et un balayage lent et périodique des divers canaux permet un enregistrement à long terme des conditions pour chaque canal présentant un comptage accru.

Dans le cas de Sizewell, le système par groupes fonctionne d'une façon continue bien que les données du détecteur de gaz de fission à collections électrostatique soient présentées en tant que comptage intégré sur une période d'environ une minute. Ceci réduit le temps pris pour confirmer l'existence d'une surimpulsion et élimine la nécessité d'un système avec sélecteur, améliorant ainsi la sûreté du fonctionnement.

L'équipement de manutention du combustible à Sizewell, diffère de celui des centrales précédentes en ce que les canaux de chargement restent installés de façon permanente dans le réacteur et que les tubes-guides au-dessus du cœur amènent les sorties de canaux dans une configuration convenant pour un seul canal de changement. Cette disposition permet d'installer des vannes de sélection démontables près des points d'échantillonnage. Ces vannes fournissent des échantillons par groupe assurant la sélection pour l'échantillonnage d'un seul canal.

Le moniteur de groupe est prévu avec dispositif d'alarme pour les comptages élevés, et c'est seulement dans le cas d'une telle alarme que les données de l'intégrateur sont enregistrées. Les données du balayage d'un seul canal sont enregistrées sur cartes perforées pour analyse ultérieure par calculateur. Lorsque le système est utilisé pour la localisation de rupture, le comptage est immédiatement à la disposition de l'ingénieur du réacteur. Des dispositifs pour l'examen de canaux particuliers sont prévus, à raison d'un dispositif pour 1 000 canaux de combustible. Un système global de surveillance des gaz est également prévu pour compléter les autres systèmes, particulièrement lorsque l'un des canaux n'est pas suivi par un système d'alarme.

On a tiré profit de l'existence d'un dispositif de détection à cellules multiples mis au point pour le réacteur AGR de Windscale. Le fonctionnement du système par groupe a été dicté en grande partie par ce dispositif et par la nécessité de maintenir de faibles débits de gaz à échantillonner en raison du très

grand nombre d'échantillons à prendre simultanément.

Les principaux organes d'équipement mis au point spécialement pour les réacteurs de Sizewell ont été les vannes primaires et secondaires de sélection: les vannes primaires doivent fonctionner dans des conditions de gaz chaud en réacteur avec une fiabilité considérable et un minimum de deux années entre les périodes d'entretien. La vanne secondaire a été mise au point à partir de la vanne primaire, mais est soumise à des conditions beaucoup moins rigoureuses.

Un perfectionnement du système de Sizewell consiste en l'adjonction d'un thermocouple dans la vanne primaire de sélection, afin de mesurer la température des gaz à la sortie de chaque canal.

A/142 Соединенное Королевство

Разработка систем обнаружения поврежденных топливных блоков атомной электростанции в Сайзуэлле

Б. Холмс, А. Д. Бутройд

Эксплуатация реакторов в Колдер-Холле и Чэпел-Кроссе показала, что для обнаружения и нахождения мест сильного и быстрого разрыва и повреждений в результате образования пустот необходимо сконструировать системы обнаружения поврежденных топливных блоков. Необходимость в быстром обнаружении слоев быстрого разрыва блоков придает особенно важное значение частоте сканирования за счет чувствительности. С другой стороны, для нахождения каналов, содержащих повреждения в результате образования пустот (что приводит к медленному нарастанию фона), требуется производить сканирование реактора с высокой чувствительностью.

Для современных реакторов была принята система двойного сканирования. В этой системе быстрое сканирование групп, содержащих до 32 каналов, обеспечивает регулярную эксплуатационную защиту, а медленным периодическим сканированием отдельных каналов производится долговременная запись показаний для каждого канала, по которой можно определить повышение радиоактивного счета.

В случае реактора в Сайзуэлле групповая система сканирования является непрерывно действующей, хотя результат на электрофильтре выражается числом импульсов, интегрированных за период примерно в одну минуту. Эта система позволяет уменьшить затраты времени на подтверждение наличия разрыва блока и устраняет необходимость в селекторной системе. Благодаря этому повышается надежность эксплуатации реактора.

Оборудование для перезагрузки топлива на реакторе в Сайзуэлле отличается от подобного оборудования, установленного на ранее построенных атомных электростанциях, тем, что загрузочные желоба остаются постоянно установленными в реакторе, а направляющие трубы, расположенные над активной зоной, имеют выпускные отверстия в форме, соответствующей простому желобу. Это приспособление позволяет устанавливать вентили селектора, снабженные дистанционным управлением, в точках отбора проб. С помощью этих вентилей производится групповой отбор проб. Вентили работают в качестве селекторов для отбора проб из отдельного канала.

Групповой дозиметрический прибор снабжается сигнальными устройствами, рассчитанными на высокий радиоактивный счет. Единственным результатом работы такого сигнализирующего устройства является выдача данных, записываемых интегратором. Данные от сканирования отдельного канала записываются на перфорированных карточках для последующего анализа с помощью электронно-вычислительной машины. Когда система используется для обнаружения места разрыва, инженер-реакторщик может непосредственно наблюдать результаты счета. Количество установок, необходимое для блокировки отдельных каналов, определялось из расчета — одна установка на каждые 1000 топливных каналов. Общая система дозиметрии газа дополняется также другими системами, особенно когда какой-либо канал не контролируется системой сигнализации.

Преимущество системы заключается в использовании многоячеистого электрофильтра, разработанного для усовершенствованного реактора с газовым охлаждением в Виндскейле. Рабочие характеристики групповой системы сканирования определялись главным образом с помощью установки и необходимостью поддерживать небольшие потоки для отбора проб ввиду очень большого количества проб, отбираемых одновременно.

Основное оборудование, разработанное специально для реакторов в Сайзуэлле, включало первичные и вторичные селекторные вентили. Это оборудование должно было работать в условиях горячего газа в реакторе со значительной степенью надежности и минимальным периодом между ремонтными работами в 2 года. Вторичный вентиль был разработан на основе первичного вентиля, но был рассчитан для менее трудных условий.

Разработка системы сканирования для реактора в Сайзуэлле включает использование термопары в первичном селекторном вентиле, позволяющей измерять температуру выходящего газа из каждого канала.

A/142 Reino Unido

Desarrollo de sistemas de detección de rotura de vainas con especial referencia a la central nuclear de Sizewell

por B. Holmes y A. D. Boothroyd

El funcionamiento de los reactores de Calder Hall y Chapelcross ha demostrado que los sistemas de detección de rotura de vainas deben proyectarse en forma a detectar y localizar reventones súbitos y frecuentes así como también fallos de cavitación. La necesidad de una detección inmediata de reventones súbitos obliga a fijar la atención en la frecuencia de exploración en detrimento de la sensibilidad. Por otro lado, la necesidad de localizar los canales en los que hayan tenido lugar fallos de cavitación, lo que se traduce en una elevación lenta de la activación de fondo, exige una capacidad de exploración muy sensible del reactor.

Para los últimos reactores se ha adoptado un sistema de exploración doble, en el que una exploración rápida, por grupos de hasta 32 canales, proporciona la protección operativa rutinaria, y otra exploración, lenta y periódica, por canales individuales, hace posible el registro a largo plazo de cada canal, para poder determinar un incremento de actividad en el mismo.

En el caso de la central de Sizewell, el sistema de exploración por grupos es continuo, aunque el resultado de salida del precipitador se presenta como una actividad integrada sobre un período de 1 minuto aproximadamente. De esta manera se reduce el tiempo preciso para confirmar la existencia de un reventón y desaparece la necesidad de un sistema selector, aumentando así la seguridad de funcionamiento.

El equipo de recarga en el reactor de Sizewell se diferencia del de las primeras centrales en que quedan permanentemente instalados en el reactor conductos de carga, y que tubos-guía por encima del núcleo dan lugar a que las salidas de los canales adopten una configuración apropiada a un conducto simple. Esta disposición facilita la instalación de válvulas retráctiles selectoras junto a los puntos de toma de muestras. Estas válvulas multiplican las tomas de muestras por grupos y funcionan como selectores para un rastreo canal por canal.

El monitor de grupo está dotado de dispositivo de alarma por actividad en exceso y únicamente se registran las salidas del integrador después que tal alarma se haya recibido. Los datos recogidos en la exploración canal por canal se registran en tarjetas perforadas para su análisis posterior en una calculadora. Cuando el sistema se utiliza para localización de reventones, la información aparece inmediatamente disponible para el facultativo responsable del reactor. Se han previsto instalaciones para bloquear canales individuales, en número de una por cada 1000 canales de combustible. Se ha previsto también un sistema monitor de gases, para reforzar otros sistemas, en particular en el caso que ningún canal sea inspeccionado por un sistema dotado de alarma.

Se ha aprovechado la existencia de una unidad de precipitación tipo multicélula desarrollada para un reactor AGR de Windscale. Las características del sistema de grupo han sido impuestas, en gran medida, por esta unidad y por la necesidad de mantener reducidos los flujos de las tomas de muestra a la vista del elevado número de muestras tomadas simultáneamente.

Las partes más destacadas del equipo desarrollado especialmente para los reactores de Sizewell han sido las válvulas selectoras primarias y secundarias. Las primeras se requieron para funcionar en las condiciones impuestas por el gas sobrecalentado del reactor con seguridad considerable de funcionamiento y durante períodos de dos años como mínimo entre períodos de revisión. La válvula secundaria se desarrolló a partir de la primaria, pero en condiciones no tan severas.

El desarrollo del sistema de Sisewell incluye un termopar en la válvula selectora primaria para facilitar la medida de la temperatura del gas de salida para cada canal.

Control of gas-borne activity arising from reactor faults

By R. D. Collins * and A. E. J. Eggleton **

The generation of electrical power by nuclear means offers many advantages, one of which, the virtual absence of atmospheric pollution arising from the power station during normal operation, is particularly desirable in well populated highly industrialised countries where atmospheric pollution is frequently a real problem. Nevertheless many such countries at present impose restrictions on the siting of nuclear power stations, as a precautionary measure against the remote possibility of a release of moderately large amounts of fission products following certain types of postulated reactor accidents. However an important and ascertainable factor is the ability of containment and gas treatment plant to trap the activity which could be released from reactor fuel. Containment and gas treatment are complementary, and both can contribute to the attainment of confidence in the safety of reactors [1]. The experiments described in this paper were undertaken to investigate the nature of fission-product releases and to establish the effectiveness of fission-product trapping systems for gas-cooled reactors. Although the greater part of the work was undertaken to provide information in support of advanced gas-cooled reactor safety studies, many of the results are of more general validity and should be applicable to the behaviour of other reactor systems.

Examination of the hazards from various fission products [2] shows that iodine is the most important on account of its high fission yield, preferential release from over-heated fuel [3], and low maximum permissible limits. Moreover, experience has shown [4–6] that it is particularly difficult to remove iodine efficiently from a gas stream as it can exist in a variety of gaseous forms in addition to being attached to particulate matter. Consequently, in this investigation penetration of the fission-product trapping system by iodine was chosen as the measure of efficiency although other fission products were also studied.

The efficiency required of a trapping system depends to some extent on the reactor type, its detailed design and the range of possible accidents. Although assessment of the safety of current reactor designs does not require such a high efficiency, an iodine penetration of 1 part in 10^4 was chosen as a target. Beyond this level it is difficult to prevent the release to the environment, by leakage from the containment vessel, becoming greater than that through the trapping system and further improvement would be of little value.

Since the chemical and physical nature of iodine depends on the conditions at the time of release and subsequently, it was considered necessary in this investigation to use conditions resembling as closely as possible those which might occur in an advanced gas-cooled reactor.

NATURE OF FISSION PRODUCTS RELEASED FROM IRRADIATED FUEL [7]

Miniature fuel elements, otherwise similar to those used in an advanced gas-cooled reactor, consisting of 25 g of uranium dioxide canned in stainless steel and irradiated for up to 2 000 MWd/t, were mounted in a furnace tube surrounded by the induction coil of a radiofrequency heater. The temperature of the fuel elements was raised to 1 500–1 600 °C for 20 minutes in a filtered mixture of either carbon dioxide and carbon monoxide or carbon dioxide and air and the released fission products were carried through a cyclone to the trapping system described later. Samples of the gas stream were taken at a number of points both before and after the trapping system using a thermal precipitator for particulate samples, an evacuated flask for total gas activity, and a sample pack of the type devised by May [8] for the various forms of iodine. The May pack contained, in order, copper gauzes for the collection of elemental iodine, a membrane filter for collection of particulate iodine, and charcoal-loaded paper backed by activated charcoal for penetrating iodine compounds.

Both in CO_2/CO and CO_2/air the proportion of fission products released from the fuel was considerably greater than that found by Parker [3] for releases in an inert gas. Between 30% and 50% of the iodine, tellurium and caesium were released but less than 0.2% of the strontium. The barium released was less than 3%, although it reached 50% in one experiment in which the fuel temperature exceeded 2 000 °C. Substantial amounts of krypton and xenon were released but not measured. Other fission products were

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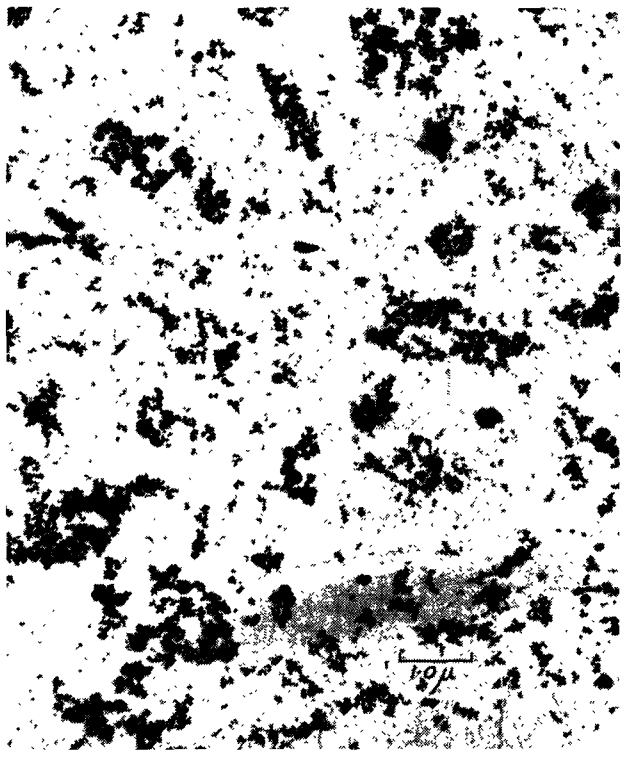


Figure 1. Typical aerosol from irradiated fuel

detected but amounts were not generally sufficient to be measured.

As the fuel can melted a dense particulate cloud developed, some of which deposited in the furnace tube and the catch pot of the cyclone. The cyclone prevented particles larger than 2 μ from reaching the thermal precipitators and blocking the inlet tubes. The precipitator samples consisted of primary particles ranging in diameter from 0.005 to 0.1 μ, agglomerated to clusters of from 0.01 to 1 μ (Fig. 1). The mean frequency size of the primary particles tended to decrease as irradiation and temperature increased, and in separate experiments with unirradiated fuel the mean frequency size of the primary particles produced in gas at a pressure of 35 atm was 2–3 times that at 1 atm.

The composition of the iodine activity, as determined by the May pack samples, was found to depend both on the nature of the gas mixture used and on the mass concentration of iodine in the gas phase which varied with the amount of irradiation the fuel had received. In the CO_2/CO mixtures none of the iodine was found in the elemental form and the amount in the form of penetrating compounds varied from about 25% at a mass concentration of 10^{-6}g/m^3 to 10% at 10^{-3} g/m^3, the remainder being attached to particles. On the other hand, when gas mixtures containing air were used, about 60% of the iodine was present in the elemental form with reduced amounts of particulate and penetrating forms. Other experiments with trace-irradiated fuel in unsealed cans showed that as little as 1% of air in the gas mixture produced a substantial amount of elemental iodine.

In order to be able to predict the deposition behaviour of the iodine in a reactor accident, samples of stainless steel, graphite, concrete and mild steel were exposed to the gas-borne activity and the rate of deposition of iodine on them measured under various conditions. In the absence of air and hence also of elemental iodine deposition velocities were as low as 10^{-3}cm/s and even on graphite at 400 °C the value was only 10^{-2}cm/s. On the other hand when air and thus elemental iodine were present, a value of 1 cm/s was measured on mild steel at 300 °C which was reduced to 0.1 cm/s by coating the mild steel with a graphite loaded paint baked on at 400 °C.

DESIGN AND PERFORMANCE OF THE TRAPPING SYSTEM

The trapping system consisted basically of a high efficiency filter for the removal of particulate activity followed by a charcoal bed for the removal of gaseous iodine. It was necessary to filter the gas before, rather than after, it passed through the charcoal bed so that gaseous iodine released by radioactive decay of particulate tellurium aerosols held on the filter could be trapped by the charcoal. In a reactor the gases reaching the trapping system can be cooled below 100 °C so this temperature was chosen as being the most stringent condition required for testing the charcoal bed. Since the high efficiency filter had a relatively low capacity for particulate matter a coarser prefilter was also used to prevent clogging. A further high-efficiency filter with associated prefilter was placed after the charcoal bed to collect any particles of charcoal containing adsorbed iodine dislodged from the bed. The complete arrangement was thus as follows:

(*a*) A glass fibre prefilter;

(*b*) A high efficiency filter of glass fibre paper through which the face velocity of the gas was 5 cm/s;

(*c*) A bed of coconut charcoal 1.4—2.0 mm mesh size, 10 cm deep by 2.5 cm diameter operated at 100 °C and a face velocity of 70 cm/s;

(*d*) Another prefilter; and

(*e*) Another high efficiency filter.

The overall penetration of activity was determined by comparing a sample of the gas stream after the system with one taken before, using the May pack samplers. As a check, the full flow from the trapping system was passed through a larger charcoal bed, the activity of which was also measured.

By varying the irradiation of the miniature fuel elements, iodine mass concentrations in the gas entering the trapping system ranging from 10^{-9} g to 10^{-3}g/m^3

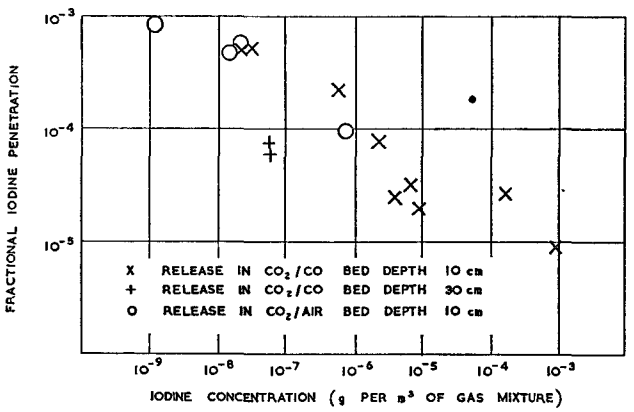


Figure 2. Penetration of trapping system by iodine vs. concentration

were achieved and Fig. 2 shows the iodine penetration observed. This decreased from about 1 part in 10^3 at 10^{-9}g/m^3 to 1 part in 10^5 at 10^{-3}g/m^3 and in the range of most interest in reactor releases (10^{-6}g/m^3-10^{-3}g/m^3) was consistently less than the target of 1 part in 10^4. The presence of air had little effect on penetration but increasing the depth of the charcoal bed to 30 cm produced a moderate improvement. In some experiments clean gas was passed through the bed for up to 100 hours after the release of activity without greatly increasing the total penetration observed.

The penetration of fission products other than iodine was usually below the limit of detection and even in the experiment in which 50% of the barium in the fuel was released, barium was not measurable in the effluent. In order to measure the penetration of long-lived fission products one experiment was performed using fuel irradiated to over 1 500 MWd/t and cooled for more than six months. Penetration of caesium was 2 parts in 10^5 and strontium less than 2 parts in 10^3. The non-gaseous fission products, with the occasional exception of a little barium, were normally all retained on the first two filters of the trapping system (*a*) and (*b*) and little activity was found on filters (*d*) and (*e*). However in one experiment filter (*b*), which was overloaded, ruptured and a considerable amount of mixed activity reached the last filter (*e*) but the overall penetration of iodine and other fission products remained less than 1 part in 10^3.

In order to discover whether fission-product iodine freshly released from irradiated UO_2 had special properties in respect of its penetrating ability, a comparison with iodine generated artificially was performed. This also permitted the range of conditions employed to be extended without the difficulty of using irradiated fuel. Separated ^{131}I mixed with an appropriate amount of stable iodine was dissolved in carbon disulphide and a drop of the solution placed in an empty fuel can. The solvent was evaporated leaving the iodine deposited on the surface, an unirradiated fuel pellet was placed in the can, and the lid was then replaced. On heating the can above its melting ponti a release of iodine occurred that was similar to that from irradiated fuel. Up to at least a concentration of 1.5×10^{-5}g/m^3 neither the May pack sampler nor the performance of the trapping system showed any difference between iodine freshly released from fuel and that released by this method. The proportions of elemental and particulate iodine released showed the same dependence on gas composition and an analysis of the penetrating compounds by gas chromatography indicated a similar composition to the iodine compounds released from fuel.

IDENTIFICATION OF PENETRATING IODINE COMPOUNDS

The chemical identification of the compounds which make iodine difficult to trap was first successfully achieved [9] with iodine generated by heating carrier-free Na^{131}I in purified air at 400 °C in a thoroughly cleaned glass apparatus. Elemental and particulate iodine were removed by freshly cleaned copper knit-mesh and a high-efficiency filter respectively, leaving about 1% of the total activity remaining in the form of iodine compounds. By solvent extraction these were divided into two fractions; one (A) readily soluble in benzene and relatively unreactive, the other (B) roughly equally soluble in benzene and aqueous solutions. The behaviour of fraction B on treatment with various reagents was complex and the identity of this fraction is still unknown. Gas chromatography of fraction A on a dinonyl phthalate column indicated the presence of 7 to 9 components, of which the first component eluted comprised about 85% of fraction A. The retention volume of this component was found to correspond within 0.02% with that of methyl iodine by a new method [10] and its molecular weight, determined by an effusion technique was found to be 152 ± 3 compared with the theoretical value for methyl iodine ($CH_3{}^{131}I$) of 146 [9]. Similarly, the second component eluted from the column was identified as ethyl iodine and with somewhat less certainty the remaining components of this fraction were identified as higher alkyl iodides up to n-butyl iodine.

Using the same techniques an analysis was performed on a sample of penetrating compounds produced in one of the experiments described earlier in which uranium dioxide fuel irradiated to 100 MWd/t was heated in a CO_2/CO mixture. The total iodine release corresponded to a concentration in the gas-phase of 10^{-5}g/m^3 and after separation of the particulate fraction, the remaining penetrating fraction released from the fuel was analysed by gas chromatography. There were two iodine activity peaks, the retentions of which corresponded exactly with those of methyl and ethyl iodides respectively, the ethyl iodide activity comprising about 1% of that of the methyl iodide. The

molecular weight of the methyl iodide fraction was found to be 145±2, close to the theoretical value of 146. No significant amount of activity was found with properties similar to those of fraction B.

Methyl iodide was also found in the penetrating compounds produced by the artificial method of simulating iodine releases from fuel described earlier and in all cases where penetrating compounds have been investigated, methyl iodide has been identified as a major component.

HIGHLY ACTIVE TEST

Certain features of an actual reactor release, such as gas temperature and pressure, and the fission product environment after release, were not reproduced in the previous experiments. It was considered desirable, therefore, to carry out at least one test in which both the conditions and the operating procedure resembled as closely as possible those following a reactor release.

A mild steel pressure vessel, volume 80, representing the reactor pressure vessel, was connected by suitable valves and pipework to a scaled trapping system similar to that described previously containing charcoal at 100 °C and thence to mild steel vessels, total volume 1 400 l, representing the reactor containment volume. Gas from the containment volume could be recirculated through the pressure vessel or through the trapping system by means of a pump. All the metal surfaces exposed to the gas were treated with a graphite loaded paint baked on at 400 °C which is sometimes employed to protect the inner surface of a reactor pressure vessel during construction.

Steel clad uranium dioxide fuel (800 g) irradiated to 2 000 MWd/t and with a total activity of 4 000 C was placed in an open graphite crucible within the pressure vessel filled with a mixture of CO_2 and CO at 400 °C and a pressure of 25 atm. The following sequence of operations, each corresponding to a possible accident condition in a reactor system was carried out.

(*a*) *Fission product release from fuel.* The fuel was heated to 1 600–2 000 °C by an induction heater during the operations (*a*) to (*e*).

(*b*) *Immediate blowdown.* After 14 min some gas containing the freshly produced fission products was discharged through the trapping system to the containment volume until the pressure had fallen to 17 atm.

(*c*) *Leakage to the containment.* Further gas was discharged to the containment, this time without passing through the trapping system, until the pressure had fallen to 12 atm.

(*d*) *Delayed blowdown.* After the fission products in the pressure vessel had been allowed to age for a further hour at approximately 400 °C the remainder of the gas was discharged through the trapping system to the containment volume, reaching a final pressure of 1 atm.

(*e*) *Recirculation.* The gas and fission products in the containment volume (previously filled with a mixture of 90% CO_2 and 10% air), were now recirculated for 1¼ hours through the trapping system. Periodically they were also circulated through the pressure vessel to examine the effect of heating the fuel in a gas mixture containing air.

(*f*) *Clean-up.* The induction heater was switched off, the pressure vessel isolated and recirculation continued through the containment vessel and trapping system for a further 20 hours to represent the condition of a pressure circuit being purged with CO_2 to exclude air from entering a break.

The only activity detected, apart from rare gases, penetrating the trapping system was iodine, the amounts and composition of which are shown in Table 1.

Table 1

Operation	Iodine entering trapping system		Total iodine leaving trapping system
	Particulate fraction	Gaseous fraction	
Blowdown	500 mc	50 mc	1 mc
Recirculation	1 000 mc	1 000 mc	100 mc
Clean-up	—	8 000 mc	900 mc

The fuel contained a total of 20 mg iodine and 200 curies of ^{131}I

Little iodine reached the trapping system during the blowdown operations and the majority arrived during the 20 hour clean-up period when the pressure vessel containing the fuel was isolated. It must therefore have been adsorbed on the walls of the containment vessel and the pipework during the blowdown period, and desorbed again later.

Clearly the penetration of the trapping system by iodine was greater than in the earlier experiments, as much as 10% escaping collection. A sample of this fraction was examined by gas chromatography and found to contain at least four components, two of which corresponded to methyl and ethyl iodide respectively and two, not previously observed, which may have been due to unsaturated aliphatic iodides. Since the major component was again methyl iodide, it was clear than an investigation of its formation and behaviour was needed to understand the unexpected results of the highly active test.

FORMATION AND BEHAVIOUR OF METHYL IODIDE

Methyl iodide was readily formed from carrier-free ^{131}I. Evaporation of an organic or aqueous solution of iodine on a clean glass or stainless steel surface left

a deposit which, when heated, gave off methyl iodide. Iodine adsorbed on to particulate matter from heated fuel and collected on a stainless steel filter also yielded methyl iodide on heating to 500 °C. The yield of methyl iodide by these methods dropped from about 50% to insignificant amounts when as little as 10 μg of carrier iodine was added. However it was found that larger amounts were produced when 3 mg of elemental iodine in a stream of CO_2 was passed through 4 m of mild steel pipe coated internally with the graphite-loaded paint used in the highly active test. Half the iodine was retained and the remainder was converted to a form which penetrated to the charcoal paper of a May pack. When the pipe was heated to 250 °C for 30 minutes a mixture of penetrating compounds containing at least 10 μg of methyl iodide was produced.

A study wasmade of the behaviour of methyl iodide in a 2 m^3 pressure vessel similar in design to that used in the highly active test and also coated with graphite paint. In a CO_2/CO gas mixture and with the pressure vessel walls at 400 °C the total gas-phase activity following an injection of labelled methyl iodide declined rapidly with a half-life of 10–20 min. When air was added, elemental iodine desorbed from the walls and appeared in the gas-phase, only to disappear again as the oxygen reacted with the carbon monoxide present. This cycle could be repeated several times but if sufficient air was added to oxidise all the carbon monoxide and leave an excess of oxygen, the amount of gas-phase activity rose and stayed constant for long periods. With the pressure vessel at 300 °C the initial decomposition of the methyl iodide took place more slowly (half-life 150 min) and at 200 °C no decomposition was observed. Addition of excess air more than doubled the decomposition rate of methyl iodide at 300 °C, and 5% of the total iodine rapidly appeared in the gas-phase as elemental iodine.

From these experiments the iodine behaviour in the highly active test can be reconstructed. During the initial heating of the fuel in the CO_2/CO mixture the usual proportions of particulate and penetrating iodine compounds were produced. Since the pressure vessel was hot, most of the iodine compounds decomposed and absorbed on to particulate matter and the walls of the pressure vessel. Because of the high graphite temperature, a dense particulate fume developed leading to agglomeration and rapid deposition, consequently little iodine reached the trapping system during blowdown and that which did was largely particulate. During the recirculation period, the air introduced into the pressure vessel caused desorption of elemental iodine and this then deposited in the cooler pipework and containment vessel. In the final clean-up period this iodine slowly desorbed in the form of iodine compounds which entered the trapping system. The capacity of charcoal to permanently retain methyl iodide is limited (see the following section) and the mixture of methyl and other organic iodides evidently overloaded the charcoal bed allowing a significant quantity to penetrate the trapping system.

METHYL IODIDE ABSORPTION

The adsorption of labelled methyl iodide on the coconut charcoal used in most of this work was studied over a range of loadings. At 100 °C, quantities smaller than 100 μg were permanently adsorbed on a 25 g bed but increasing penetration was observed as the loading was increased above this limit. At room temperature there was a longer delay before breakthrough occurred due to enhanced physical adsorption but the amount of methyl iodide permanently retained was considerably less.

The main duties of the trapping system of an advanced gas-cooled reactor are either (*a*) to depressurise the pressure circuit after a single channel melt-out or (*b*) to provide continuous treatment of relatively clean coolant after a pressure circuit failure. The trapping system is not therefore required to operate for long periods while retaining a large amount of methyl iodide and the critical loading at 100 °C, which corresponds to 4 g methyl iodide/t of charcoal, would not be unduly restrictive. But it is desirable to use the most effective absorbent for methyl iodide practicable and alternatives to coconut charcoal were considered.

Twenty or so common aqueous reagents including strong oxidising and reducing agents, soluble silver salts and amines were brought into contact with a gas mixture containing methyl iodide in a foam column for a period of 10 s. None of them produced any worthwhile decontamination at room temperature except acid potassium bromate solution which allowed a penetration of 1 part in 100. Excellent removal of methyl iodide was obtained with the solid adsorbent Hopcalite (an oxidation catalyst containing MnO_2 and CuO) but it was unsuitable for use in gas mixtures containing CO since it was reduced and lost its activity.

Attention was then directed to alternative types of charcoal. Copper and silver-impregnated charcoal showed no improvement over the standard coconut charcoal but a coal-based charcoal (Sutcliffe Speakman & Co. Ltd. type 207B) was found to have a much higher loading limit. At 100 °C a 25 g bed loaded with 300 μg of methyl iodide retained all but 2 parts in 10^5 for 12 hours and all but 1 part in 10^6 of 2 mg for 1 hour.

Most of the work with irradiated fuel described previously was performed with the coconut charcoal at 100 °C because this was thought to represent the most stringent condition. But three experiments performed at room temperature in fact showed unusually large iodine penetrations around 1 part in 10^3 and this behaviour was also found in the labelled methyl iodide experiments. In contrast, the coal-based charcoal (type 207B) maintained its improved performance

down to room temperature. Not only was a penetration as low as 1 part in 10^4 observed in irradiated fuel experiments but even when a 25 g bed was loaded with 13 mg of methyl iodide less than 1 part in 10^4 penetrated in 24 hours elution. This loading, equivalent to 500 g/t, may be taken as the permissible loading for reactor purposes.

CONCLUSIONS

Except for the rare gases and iodine, fission products are released solely in particulate form when irradiated uranium dioxide is heated in carbon dioxide containing either carbon monoxide or air. In CO_2/CO mixtures iodine is released as a mixture of particles containing iodine and volatile compounds, largely alkyl iodides of which methyl iodide is a major component. In mixtures containing air, substantial quantities of elemental iodine are also produced. The proportions of these forms produced depend on the conditions and there may also be interconversion after release. Thus elemental iodine may be adsorbed on surfaces or particulates and partially desorbed again either as elemental iodine or alkyl iodides and alkyl iodides can be decomposed by heat to elemental iodine.

There is, as yet, insufficient information to predict the proportion of each iodine form produced under given conditions. Nevertheless all forms of iodine and also the other particulate fission products can be adequately trapped by a combination of high efficiency filters and charcoal bed provided the critical loading of the bed for iodine compounds is not exceeded. The critical loading depends on the type of charcoal used and the temperature of the bed. For the coal-based charcoal tested the limit is in excess of 500 g methyl iodide/t at room temperature. Some other types of absorbent may be superior under different conditions but for gas-cooled reactors charcoal appears to be the most suitable.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/186 Royaume-Uni

Fixation de la radioactivité transportée par gaz à la suite de défaillances de réacteur

par R. D. Collins et A. E. J. Eggleton

Pour assurer le captage approprié de toute activité transportée par gaz à la suite d'un accident dans un réacteur amélioré refroidi par un gaz (AGR), le système de fixation préféré (composé d'un adsorbeur au charbon de bois et de filtres de haute efficacité) a été essayé sur toute la gamme des conditions possibles. ^{131}I est le produit de fission le plus dangereux et la difficulté de sa fixation provient des formes chimiques et physiques qu'il peut prendre. Le système de fixation a donc été essayé avec des produits de fission émis par des cartouches de combustible irradiées et chauffées dans différents mélanges de CO_2/CO/air. A des températures légèrement supérieures au point de fusion de la gaine, les dégagements d'iode, de tellure et de cesium sont compris entre 10 et 50%, mais les dégagements de strontium sont inférieurs à 0,2%.

On a trouvé que les principales caractéristiques du comportement de ^{131}I libéré pouvaient, par une technique appropriée, être reproduites utilement en utilisant ^{131}I isolé en laboratoire. Une proportion considérable est transportée avec les vapeurs, mais on n'a détecté aucune pénétration du filtre absolu. On a trouvé des composés vaporisés qui ne se déposent pas rapidement sur les surfaces de cuivre et les plus pénétrants de ceux-ci ont été identifiés comme des iodures de méthyle et d'autres alkyles, précédemment trouvés avec l'iode isolé sans porteur, dilué avec de l'air propre. La transformation des différentes formes a été étudiée dans la phase gazeuse et sur des surfaces.

On a trouvé que le charbon de bois retarde l'iodure de méthyle par adsorption physique, mais ne le retient

d'une façon permanente que si la charge se trouve inférieure à une certaine limite. Pour la matière normale (choisie pour sa bonne rétention des autres formes) la limite était d'environ 1 μg/g de charbon de bois à 100 °C, et moindre à la température du laboratoire, mais on a constaté que d'autres charbons de bois ont une limite supérieure, même à la température du laboratoire.

On a trouvé que des produits de fission fraîchement libérés à des concentrations de 10^{-3} à 10^{+3} μg/m³ pénétrant un système de fixation comprenant une couche de charbon de bois de 10 cm. Cette pénétration varie de 1 pour 10^3 pour la concentration inférieure, à 1 pour 10^5 à la concentration supérieure, et l'iodure de méthyle forme une proportion considérable de l'activité gazeuse.

La seule pénétration sérieuse de produits de fission s'est produite lors d'une expérience dans laquelle de grandes quantités de radioactivité ont été libérées dans des conditions favorisant la formation de composés. L'élution prolongée de la couche fortement chargée a eu pour résultat la pénétration par l'activité, due principalement aux iodures d'alkyles. La charge totale a été 10 fois plus grande que la limite pour l'iodure de méthyle du charbon de bois utilisé, mais se trouvait dans les limites du fonctionnement satisfaisant de certains charbons de bois essayés depuis.

A/186 Соединенное Королевство

Контроль радиоактивности газов, возникающей в случае неисправности реакторов

Р. Д. Коллинз, А. Э. Дж. Эгглтон

Для того чтобы убедиться, что любая радиоактивность в газе, возникшая в случае аварии реактора типа AGR, будет соответствующим образом уловлена, в широком диапазоне возможных условий была испытана улавливающая система, состоящая из угольного поглотителя и фильтров высокой эффективности. Наиболее опасным продуктом деления является J^{131}; трудности его улавливания связаны с химической и физической формами, которые он может принимать. Поэтому для испытания системы улавливания использовали продукты деления, образующиеся в небольшой трубке с облученным топливом, нагреваемой в различных смесях двуокиси углерода, окиси углерода и воздуха. При температурах несколько выше точки плавления оболочки выделяются иод, теллур и цезий в количестве от 10 до 50%, а выделение стронция было меньше 0,2%.

Было установлено, что главные особенности поведения J^{131} при выделении могут быть при помощи соответствующих методов полезным образом воспроизведены с использованием разделенного J^{131}. Значительное количество было добавлено в дым, однако проникновение через абсолютные фильтры установлено не было. Были найдены соединения в виде паров, которые не осаждаются быстро на медных поверхностях, а наиболее проникающие из них были идентифицированы в виде иодистого метила и других иодистых алкилов, которые были найдены ранее в разделенном свободном от носителя иоде, разбавленном чистым воздухом. Было изучено взаимное превращение форм в газовой фазе и на поверхности.

Установлено, что древесный уголь замедляет физическое поглощение иодистого метила только до определенного постоянного уровня, если нагрузка была ниже этого уровня. Для стандартного материала (выбранного по хорошим удерживающим свойствам по отношению к другим формам) был выбран предел около 1 *мкг/г* древесного угля при температуре 100°С и меньше при комнатной температуре, но найдены некоторые сорта угля с более высоким пределом даже при комнатной температуре.

Свежевыделившиеся продукты деления при концентрациях 10^{-3} — 10^3 *мкг/м*³ проникают через систему улавливания, включая слой древесного угля толщиной 10 *см*, в количестве от одной тысячной доли при малых концентрациях до одной стотысячной при более высоких концентрациях, причем значительную долю газообразной активности составляет иодистый метил.

Единственный случай серьезного проникновения продуктов деления наблюдался в эксперименте, в котором большие количества активности выделились в условиях, благоприятных для образования соединений. Продолжительное вымывание сильно загруженного слоя привело к проникновению активности иода, идентифицированной главным образом в виде иодистых алкилов. Общая нагрузка была на порядок величины выше предела для иодистого метила при использованном сорте угля, но лежала в пределах, обеспечивающих удовлетворительную работу других испытанных сортов угля.

A/186 Reino Unido

Control de actividad de gases escapados accidentalmente de un reactor

por R. D. Collins y A. E. J. Eggleton

La memoria trata de los ensayos realizados y de los resultados obtenidos con un sistema colector de los gases activos que puedan escaparse de un reactor del tipo AGR como consecuencia de un accidente. El sistema utiliza carbón vegetal como material adsorbente y filtros de elevado rendimiento, llevándose a cabo los ensayos según una gama de condiciones consideradas como las más probables. El yodo-131

constituye el producto de fisión más peligroso y la mayor o menor dificultad de atraparlo depende de su forma física y química. Para los ensayos se utilizaron los productos de fisión procedentes de barritas de combustible irradiado que se calentaron en presencia de diferentes mezclas de CO_2, CO y aire. A temperaturas ligeramente superiores al punto de fusión de la vaina, se obtuvo una proporción de yodo, teluro y cesio comprendida entre el 10 y el 50%, mientras que la del estroncio no llegó al 0,2%.

Los ensayos indicaron que las principales características del yodo-131 desprendido como producto de fisión podían, mediante una técnica apropiada, reproducirse corectamente utilizando yodo-131 separado en el laboratorio. Transportado en considerable proporción con los humos, no se observó que atravesase el filtro absolute. Por otra parte se observaron compuestos gaseosos que no se depositaban rápidamente sobre superficies de cobre y se identificaron, como los de mayor capacidad de penetración de filtro, los yoduros de metilo y otros yoduros alquílicos que ya habían sido previamente hallados en yodo aislado sin portador, diluido en aire puro. El estudio de la conversión de unas formas en otras se realizó tanto sobre superficies como en la fase gaseosa.

La retención por adsorción del yoduro de metilo por el carbón vegetal sólo resultó permanente si la cantidad de aquél no rebasaba ciertos límites. Para el material que se venia utilizando normalmente (escogido en razón de su buena capacidad de retención de otras formas), el límite era de 1 μg/g de carbón a 100 °C y menos a la temperatura ambiente, pero se han encontrado otros carbones vegetales con un límite más alto de adsorción incluso a la temperatura ambiente.

Las observaciones de productos de fisión recién liberados, a cuya actividad el yoduro de metilo contribuía en considerable proporción, en concentraciones de 10^{-3} a 10^{+3} μg/m^3, indicaron la penetración a través de un lecho de carbón vegetal de 10 cm de cantidades que oscilaban desde una parte en 10^3 para la concentración inferior, hasta una parte en 10^5 para la superior.

El único caso de penetración seria de un producto de fisión tuvo lugar en un experimento en el que una gran proporción de actividad fue liberada en condiciones que favorecían la formación de compuestos, y la prolongada elución del lecho excesivamente cargado se traducía en el paso de la actividad del yodo, principalmente en forma de yoduros alquílicos. La carga total era un orden de magnitud por encima del límite correspondiente al carbón vegetal que se empleaba normalmente, pero queda holgadamente dentro de los límites de utilización de algunos carbones últimamente ensayados.

US experience on the release and transport of fission products within containment systems under simulated reactor accident conditions

By W. B. Cottrell,* W. E. Browning, Jr.,* G. W. Parker,* A. W. Castelman, Jr.** and R. L. Junkins ***

A summary of recent US experience on the release and transport of fission products under simulated reactor accident conditions is presented herein, but is limited to phenomena occurring within the reactor building or containment structure. Relevant data are being obtained at over a dozen different laboratories in the US, as listed in Table 1. The support for this type of work has increased manifold over the several years and is still increasing. In addition to the specific investigations (usually short-range studies) undertaken for various reactor programs, long-range, general studies covering this entire area are sponsored by the US Atomic Energy Commission's Reactor Safety Program. Investigations sponsored in this program are, or will be, carried out in several large-scale facilities with high levels of radioactivity, as well as in laboratory-scale systems with trace levels of radioactivity.

Although the question of fission-product release — principally from metallic uranium, uranium alloy and UO_2 fuel materials — has been of concern for many years, it has been within the last few years only that any significant effort has been devoted to fission-product attrition due to transport processes (both natural and induced) within containment systems. Since such processes may be singularly dependent upon the chemical and physical form of the released fission products, the last few years have also witnessed increased concern regarding the characterization of the released fission products and their transport behavior, as well as the determination of the amount released.

EXPERIMENTAL FACILITIES

All experimental work relative to fission-product release and transport involves some means of releasing either real or simulated fission products and some means of determining fission-product release. These minimal requirements are generally supplemented by sufficient sampling to permit characterization of the released fission products and means for determining transport behavior in various systems under a wide variety of possible accident conditions. Facilities for such investigations range from laboratory-sized mock-ups to full-size experimental facilities in which containment system tests can be conducted and an actual reactor core can be melted.

Fuel melt-down techniques

The chemical and physical state of the volatilized fuel material and therefore its subsequent behavior are determined in part by the melting conditions, environment and techniques. Fission-product release investigations are principally concerned with those release techniques which are capable of causing melting or near-melting in fuel materials under realistic environmental conditions. All the more common techniques are listed in Table 2 together with the installations and principal investigators who have employed the technique.

Major facilities for release and transport studies

Most of the work on fission-product release and transport to date has been performed in small-scale laboratory-type facilities, which are ideally suited for exploring the wide range of experimental parameters. Since it is virtually impossible to simulate all accident conditions, that is, heating, materials, environment, and, particularly, geometry, the results of these laboratory-type experiments are subject to question when extrapolated to predict the expected results of a melt-down or activity release in an actual system. Furthermore, in these laboratory systems it is not possible to evaluate the effectiveness of transport phenomena and countermeasures which might exist or be applied in an actual system. Accordingly, the USAEC has embarked upon a complementary experi-

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Table 1. US fission product release and transport research

Contractor	Principal investigator	General area of investigation	Program status	Principal reports
General Electric—ANPD	M. N. Myers T. D. Roberts	Fission-product deposition	Continuing	[1,2]
Battelle Memorial Institute (BMI)	A. Abriss G. E. Raines	Fission-product deposition	Continuing	[3–5]
Brookhaven National Laboratory (BNL)	A. W. Castleman, Jr.	Characterization of released fission products	Continuing	[6–8]
Savannah River	A. H. Peters G. H. Prigge	Removal of fission products in containment enclosures	Continuing	[9–11]
Illinois Institute of Technology	J. Rosinski	Removal of fission products in containment enclosures	Completed	[12]
Harvard School of Public Health	L. Silverman	Removal of fission products in containment enclosures	Continuing	[13–15]
Hanford Atomic Products Operation	E. R. Irish R. K. Hilliard	Fission-product release and transport	Continuing	[16–19, 63]
	L. C. Schwendiman	Fission-product transport	Continuing	[20–22]
	H. W. Heacock	Fission-product confinement	Continuing	[23–25]
A. D. Little, Inc.		Fission-product removal	Completed	[26]
Mine Safety Appliance Corp. (MSA)	S. J. Rodgers J. W. Mausteller	Fission-product release	Completed	[27, 28]
Phillips Petroleum (NRTS)	T. R. Wilson J. M. Waage	SNAPTRAN	Initiating	[29–31]
	T. R. Wilson G. B. Matheney	Loss-of-flow test (LOFT)	Initiating	[32,33]
	A. A. Wasserman A. H. Spano	Power burst facility (PBF)	Proposed	[34]
Sandia Corporation	M. Cowan	Fission-product transport	Completed	[35]
Dow Chemical	P. D. Erickson	Fission-product removal	Completed	[36]
Lovelace Foundation	J. C. Gallimore T. T. Mercer	Fission-product characterization	Continuing	[37]
General Atomics	J. M. Tobin	Fission-product deposition	Completed	[38]
Oak Ridge National Laboratory (ORNL)	G. W. Parker W. E. Browning, Jr. R. A. Lorenz W. B. Cottrell L. F. Parsly	Fission-product release, characterization, transport, deposition, and removal	Continuing	[39–54][a]

[a] Material appearing in Refs. [39, 40, 41, and 42] has been included in various forms in the Transactions of the American Nuclear Society (particularly Vol. 6, No. 1), the Seventh and Eighth Air Cleaning Conferences, and the First, Second, and Third Nuclear Reactor Chemistry Conference reports.

Table 2. Comparison of fuel melt-down techniques

Melt-down technique	Installation and principal investigator	Materials melted	References	Comments
Arc image	ORNL, G. W. Parker	UO_2	[53]	Specimen size limited
Induction heating	ORNL, G. W. Parker	UO_2	[53]	Requires surrounding conducting material to heat UO_2 up to temperatures at which it becomes conductive
	Hanford, R. K. Hilliard	U metal	[18]	
	MSA, S. J. Rodgers	UO_2-BeO	[28]	
	BNL, A. W. Castleman, Jr.	U metal	[6]	
Resistance heating	ORNL, G. W. Parker	U metal, UO_2	[53, 55]	When specimen resistance heated, circuit broken at onset of melting, otherwise temperature limited by furnace materials
	Hanford, R. K. Hilliard	U metal	[56]	
Plasma heating	ORNL, W. B. Cottrell	UO_2	[49]	Has higher temperature capability than any of the above techniques
	GE-ANPD, R. K. Conn	BeO	[57]	
In-pile test	ORNL, ORR, W. E. Browning, Jr.	UO_2	[54, 58]	Nuclear heating but expensive
	ORNL, TREAT, R. A. Lorenz	UO_2	UO_2 [59]	
	Phillips, PBF		[34]	
Reactor melt-down	Phillips, LOFT, T. R. Wilson		[30, 32]	Most realistic and most expensive
	Phillips, SPERT, F. Schroeder		[60, 61]	

Table 3. Major facilities for release and transport investigations

Designation	Installation	Principal investigator	Contained volume (ft^3)	Surface-to-volume ratio (ft^{-1})	Status	References
Containment mock-up facility	ORNL	G. W. Parker	6.3	2.2	Operational	[62]
Nuclear Safety pilot plant (NSPP)	ORNL	W. B. Cottrell	1 350	0.43	Operational	[48]
Containment systems experiment (CSE)	Hanford	E. R. Irish	32 700	0.17	Being designed	[63]
Special power excursion reactor test (SPERT)	Phillips	F. Schroeder	[a]	[a]	Operational	[60, 61, 64]
Loss-of-flow test (LOFT)	Phillips	T. R. Wilson	282 000	0.07	Being designed	[32]
Power burst facility (PBF)	Phillips	A. A. Wasserman	125 000[b]	0.125[b]	Proposed	[34]

[a] Open structure; not intended for containment.
[b] Based upon preliminary design of reactor bay.

mental program involving the construction of a variety of experimental facilities which will permit both the conduct of release and transport investigations under actual reactor accident conditions and which will provide the basis for the extrapolation of the mass of existing data already available from laboratory tests. The significant facilities that constitute a part of this program are listed in Table 3.

FISSION-PRODUCT RELEASE

It is important to know the fraction of the fission-product inventory which may be released in a volatile or otherwise mobile form in a nuclear accident. Historically, probably the first fuel to be subjected to a melt-down to determine fission-product release for a hazards analysis was a uranium oxide dispersion in stainless steel [65]. Next in order were the uranium alloys [50, 66] and metallic uranium [16, 52], followed by uranium oxide [39, 53] and uranium carbide [67, 68]. The most useful of these studies were those that sought to correlate the effects of irradiation history, time at temperature, and atmosphere, and those that sought to characterize the released products by chemical state, as well as particle size. Data have been obtained on release from uranium and UO_2 as some function of most of the above parameters while limited data are also available on release from other fuels, such as BeO and UC.

Uranium metal and alloys

The release of fission products from irradiated uranium metal has been studied at temperatures up to 1 400 °C [52, 56]. Larger fractions of the available fission products are released in air atmospheres than in carbon dioxide or steam [52, 56]. An inert atmosphere, such as helium, results in a still lower release quantity, particularly for fission products which are more volatile in oxide form than in elemental form. Most of the later studies have been concerned with exposure of samples to air atmospheres as the worst case, using the time and temperature variables to control the percentage of the metal oxidized. At low burn-up, those fission products which are volatile in elemental or oxide form are released to a greater extent with increases of the fraction of uranium oxidized [56]. The effect of irradiation exposure is shown in Table 4.

The release of semi-volatile and nonvolatile elements is independent of burn-up, but volatiles are released rapidly and nearly quantitatively from highly irradiated specimens. This is attributed to the accumulation of the volatiles in bubbles and cracks in the uranium matrix

Table 4. Effect of irradiation exposure on release of fission products from metallic uranium [19]

Element	Fraction released from specimen[a]		
	10^{14} n/cm^2 [b]	10^{18} n/cm^2 [b]	4×10^{20} n/cm^2 [b]
Xenon	0.70	0.95	1.00
Iodine	0.60	0.85	0.98
Caesium	0.30	0.60	0.80
Tellurium	0.65	0.65	0.65
Strontium	0.005	0.005	0.005
Barium	0.005	0.005	0.005
Ruthenium	0.001	0.001	0.04
Molybdenum	[c]	0.001	0.04
Zirconium	0.0003	0.0003	0.0003
Cerium	0.0001	0.0001	0.0001
Neptunium	[c]	0.0005	[c]
Plutonium	[c]	[c]	0.0004
Uranium	[c]	[c]	0.0003
Uranium oxidized	0.70	0.70	0.92

[a] 11.7 g cylinders heated in air for 24 min at 1 200 °C.
[b] Exposure prior to fission-product release by induction heating.
[c] Not analysed.

during irradiation. Upon heating to 1 000 °C or greater, the bubbles expand and break through the surface. Table 4 shows that uranium oxidation increases at higher burn-ups, an effect explained by the increased surface area created by the expanding fission-gas bubbles. Ruthenium and molybdenum release also increase significantly at the higher burn-ups because of the better opportunity for oxidation to their volatile oxides.

An interesting anomaly observed in these tests was that the release of ^{133}Xe occurred as a "burst" from uranium which had been irradiated to 4×10^{20} n/cm^2 (1 340 MWd/T), while specimens irradiated to lower levels continued to release gas throughout the out-of-reactor heating test. Iodine and caesium showed similar effects, but to a lesser degree.

Loosely fitted aluminum jacketing retarded the rate of uranium oxidation in air, apparently by an alloying action [52]. Rapid penetration of the aluminum into the uranium was observed. The fission-product release rate was correspondingly suppressed. Results of investigation of fission product releases from zirconium-clad uranium-zirconium alloy in a steam atmosphere indicated release fractions generally similar to those of uranium metal fuel [69]. This was also true of uranium-aluminum alloy fuel [70, 71] in steam-air mixtures and in helium. The fission-product release temperature relative to the melting point of the alloy is the basis for the comparison of release fractions.

The percentages of release of many fission-product elements have been correlated with time, temperature, atmosphere, and other parameters. The ranges of release of various groups of isotopes are listed in Table 5 [18].

Uranium dioxide

Uranium dioxide is an attractive reactor fuel material not only because of reasonable fabrication costs but also because of its inherent safety factors of inertness to most reactor coolants, dimensional stability, even under high burn-up, high degrees of gas retention at operating temperatures, and its very high melting point, approximately 2 850 °C, which permits the conclusion in most hazards analyses that almost nothing other than a complete loss of coolant at high power density would lead to UO_2 fuel melting. The susceptibility of UO_2 to oxidation to U_3O_8 is perhaps its major weakness, but this reaction does not occur above 1 500 °C, at which temperature the boiling points of many of the fission-product elements have already been exceeded.

Table 5. Classification of fuel and fission-product elements according to volatility [18]

Class	Elements	Percentage range of release from uranium in air at 1 200 °C
Volatile	Xe, I, Cs, Te, Kr	20–100
Slightly volatile	Sr, Ba	< 1
Nonvolatile	Zr, Ce, Np, Pu, U	< 0.1
Volatile oxide	Ru, Mo	0.01–5

Fission products may be released from UO_2 by various mechanisms that could exist in an accident as diffusion, oxidation and melting. There is an extensive body of literature [72, 73, 74] on diffusional release in an inert atmosphere and several reports and articles describe the processes and data in considerable detail. At normal operating temperatures, diffusional release is very slow; however, at 1 700 °C and above a more rapid release process occurs that is related to grain growth or mass transfer. An estimation of the numerical value of fission-product diffusion into pure helium in both regions and a measure of the enhancement with degree of burn-up is given in Fig. 1. Since UO_2 is relatively inert to steam, it is generally believed that in the absence of air (or oxygen) the data on diffusion in inert gas may be taken as applicable to a steam atmosphere with reasonable confidence. Allowance must be made for the time for diffusion by estimating the percentage contained in the initial burst from time-profile studies [75].

The second process which may occur and which is fairly well characterized is oxidation in air. This may occur at any time cladding is ruptured in the presence of air at an upper limiting temperature of about 1 500 °C down to about 200 °C. Since the conversion of UO_2 to U_3O_8 involves a change in crystal form as well as density, oxidation leads to rather high fission-product release when it occurs above 800 to 1 000 °C [53], as shown in Fig. 2. Burn-up also has an effect of some significance, as shown in Fig. 3, for oxidation temperatures of 1 000 and 1 200 °C [101].

The maximum credible release from UO_2 would be expected to arise from the third and most catastrophic process: melting and vaporization. Both out-of-pile [75] and in-pile [54, 76] release studies have shown that melting produces nearly quantitative loss of the rare gases, halogens, caesium, and tellurium, as shown in Table 6, as a function of atmosphere and burn-up. Ruthenium is not released unless air is present. The alkaline earth, rare earth, and refractory elements are not significantly released, since these form solid solutions in UO_2. Transient or prompt melting [77] leads to a somewhat suppressed release, which probably increases both with reactor period and with ambient conditions that extend the length of time the material is molten.

Uranium-graphite

Considerable early work [78, 79] was done on fission-product behavior of uncoated UO_2 in a graphite

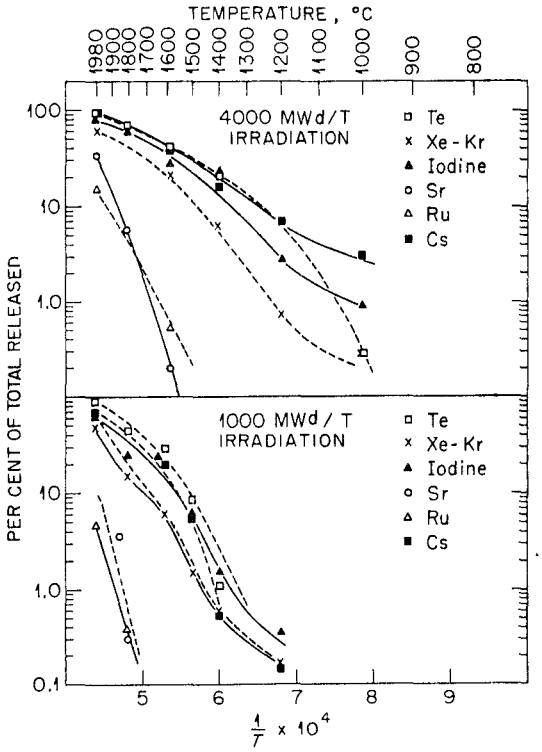


Figure 1. Release of fission products by diffusion from highly-irradiated PWR-type UO_2 heated for 5.5 hours in purified flowing helium

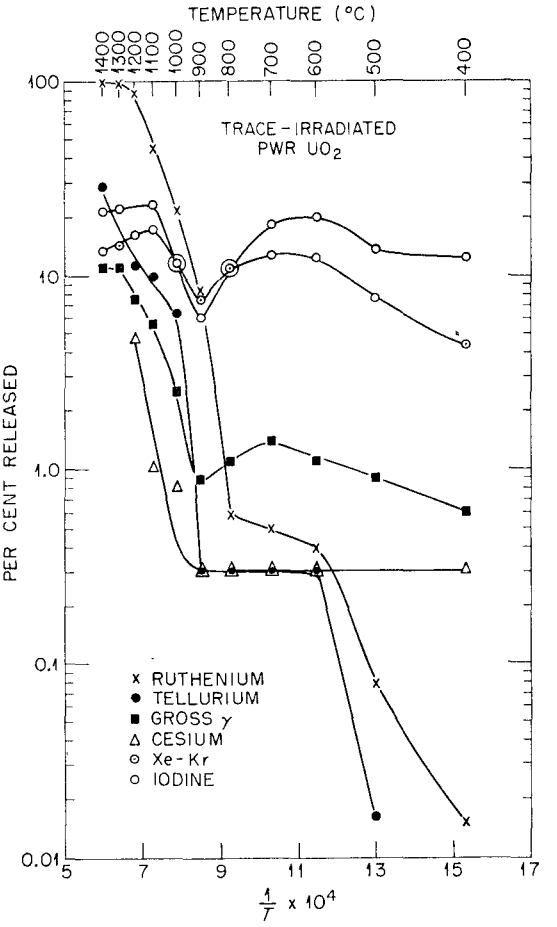


Figure 2. Fission-product release by the oxidation of UO_2 to U_3O_8 in air, showing discontinuity between 600 and 900 °C

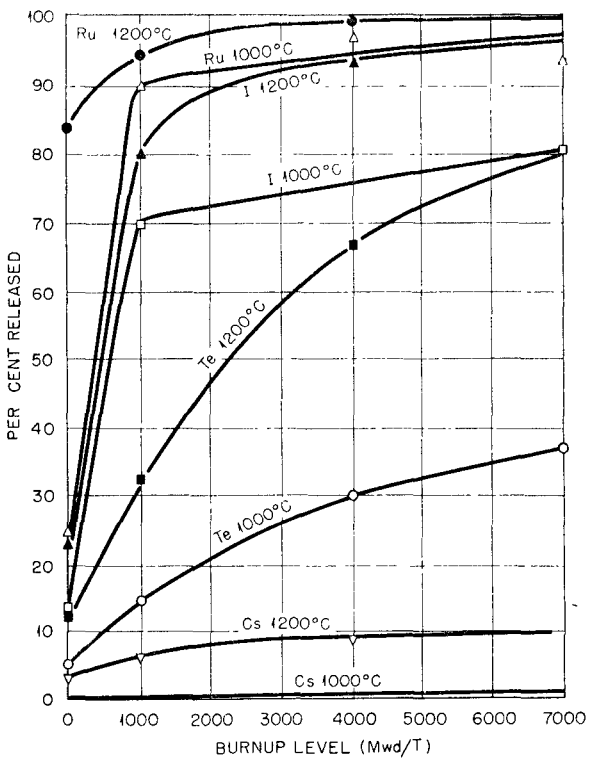


Figure 3. Effect of burn-up on fission-product release by oxidation at 1 000 and 1 200 °C

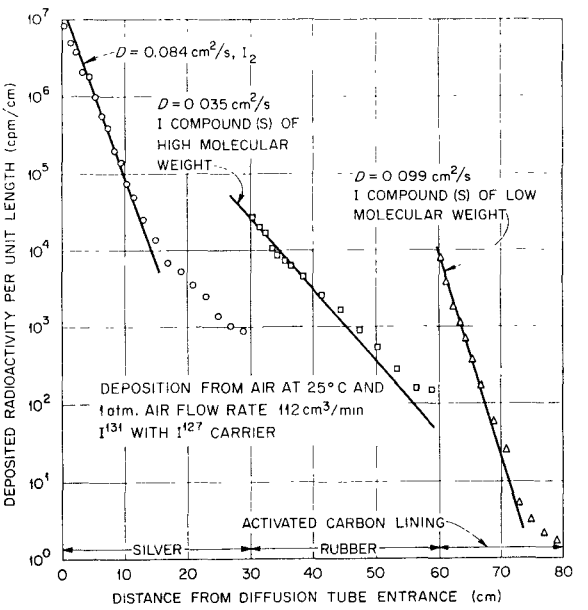


Figure 4. Diffusional deposition of elemental iodine and iodine compounds

Table 6. Effect of irradiation and atmosphere on fisssion-product release resulting from the melting of UO_2[a]

Atmosphere	Heat source	Irradiation exposure (MWd/T)	Weight of UO_2 (g)	Number results averaged	Percentage of individual fission products released [b]								UO_2 vaporized	Reference
					Xe-Kr	I	Te	Cs	Ru	Sr	Ba	Ce		
Helium														
Pure	Induction	Trace	29.0	6	97.9	91.7	95.6	71.6	0.41	0.41	3.2	0.15	~0.2	75
	In-pile	Trace	5.6[c]	5	—	99.3	96.6	98.2	76.8	60.7	68.5	53.2	—	54
						90.2[d]	74.7[d]	76.6[d]	3.78[d]	1.5[d]	1.15[d]	0.32[d]	—	—
	Resistance	Trace	39[e]	3	51	34	42	43	0.7	1.4	4.2	<0.5	0.4	75
Impure	Arc image	Trace	0.22	2	99.5	89.7	92.0	91.3	61.0	2.1	4.6	2.2	—	85
	Arc image	2 800	0.03	3	99.9	92.2	98.2	98.5	90.4	2.1	6.6	5.1	21.2	85
Air	Arc image	Trace	0.2	2	98.4	94.9	79.1	37.7	67.7	0.2	0.5	0.5	—	85
	Arc image	2 800	0.04	3	100.0	99.7	93.6	92.5	95.0	0.4	1.8	3.0	—	85
CO_2	Arc image	Trace	0.2	3	80.6	76.8	71.2	60.9	44.9	0.3	1.1	0.85	14.1	85
	Arc image	2 800	0.02	3	99.9	98.7	98.6	90.2	74.3	0.5	2.5	2.8	—	85
	Arc image	11 000	0.05	3	99.9	99.9	99.0	96.6	79.1	0.6	2.9	2.3	—	85

[a] UO_2, with oxygen-to-uranium ratio of 2.04 and density 95% of theoretical (average) except where noted.
[b] Released from fuel material except where noted.
[c] UO_2 clad in stainless steel.
[d] Released from high-temperature zone which includes fuel material and thermal insulation, minimum temperature, 1 000 °C.
[e] UO_2 density 93% of theoretical.

matrix. Although the subsequent discovery of uranium reaction and migration in graphite [8] has diminished interest in uncoated UO_2, pyrolytic-carbon-coated uranium oxide, or uranium carbide particles surrounded or embedded in graphite, are now proposed for use in various high-temperature gas-cooled reactors. Fission-product release under normal operating conditions has been shown to be acceptable for a number of material combinations [81, 82]. In an experiment in which coated UC-graphite fuel was burned in air in-pile in the Oak Ridge research reactor, more than half of all fission products (except Cs, 45%) were retained in the fuel residue when 50% of the trace-level irradiated fuel burned [76]. The release of Sr, Zr, Ba, Ce, and U from the high-temperature zone of the furnace averaged less than 2%, but large fractions (20 to 40%) of I, Te, and Cs escaped.

UO_2-BeO

The release of the more volatile fission products (I, Te, Cs, Ru, Xe, Kr) during the melting of ceramic fuel elements made up of chemically stabilized UO_2 dispersed in a BeO matrix has been determined [28, 57, 83] to approach 100% after fuel melting. The Sr and Ba releases after melting were initially only a fraction of a per cent, but they increased with time as the fuel was maintained molten. Particle measurements indicated that the major fraction of the activity was associated with released uranium.

CHARACTERIZATION OF RELEASED FISSION PRODUCTS

Characterization of the released fission products by physical and chemical forms is needed in order to predict fission-product behavior in a variety of environments, including the conditions under which the fission products would condense, adsorb on particles, react, or remain in the vapor state. Hence, considerable effort is now being directed toward this end, particularly at BNL, Hanford, and ORNL.

Particles formed from oxidation of uranium in air

The temperature at which uranium metal oxidizes in air has a pronounced effect on the size of particles transported [22]. At air velocities of 8.3 cm/s and lower, the quantity of oxide particles, primarily 0.015 to 0.5 μ diameter, collected 20 in from the sample was greatest at 1 200 °C, while at 400 to 800 °C the average particle diameter was 2 to 5 μ. Assuming that the smaller particles found at both 1 000 and 1 200 °C arose from the condensation of sublimed UO_3, a large decrease in their concentration with decreasing temperature would be predicted, as observed. As the air velocity is increased, aerodynamic entrainment of somewhat large particles overshadows release by direct volatilization. Qualitatively, samples with higher burn-up exhibit accelerated particle release [84]. Tests have shown that ^{132}Te is carried on particles with diameters greater than 200 Å, while the majority of ^{131}I is carried on particles having a bi-modal size distribution with peaks at 15 and 60 Å [37]. In other studies, also with uranium heated in air, less than 2% of the Te, Cs, Zr, and Ba was associated with particles larger than 1 μ mean diameter, while 6% of the iodine deposited with particles larger than 1 μ [19].

Particles formed from vaporization of UO_2

Significant amounts of UO_2 are vaporized at the melt temperature, although most of this is redeposited

in hot regions [85]. Most of the particles formed are 0.01 to 0.1 μ in size. Iodine, Te, Cs, and Ru are the major fission products transported out of the heated regions by air. Stainless steel cladding alters the particle size distribution [75]. Eighty-seven per cent of the particles transported from unclad UO_2 are less than 0.5 μ. In helium, a large fraction of the fission products released from stainless-steel clad UO_2 heated in-pile is found with gas-borne particles in the range 10 to 100 Å in diameter. The distribution of released fission products and uranium deposited from a gas flowing laminarly through a diffusion tube was analysed to yield information about vapors, gas-borne particles, and the amounts of released material associated with these forms [86, 90]. About 20% of the I, Te, and Cs carried out of the furnace is attached to 20 Å particles. Some Sr and Ba are also found on these particles.

Chemical state of iodine released from uranium alloys in helium and air

At temperatures between 1 150 and 1 250 °C, iodine was released from uranium and a U/3.5% Mo alloy into helium in combined form as evidenced by its condensation at 250 to 350 °C in a thermal gradient [6, 7] which is too high for condensation of elemental iodine. Similar behavior was noted for fuels irradiated over the range 3.0×10^{14} to 5.4×10^{19} n/cm². This deposit was unstable in air at 250 to 350 °C, and elemental iodine was slowly released. Uranium iodides oxidize quite readily in air to form UO_2 and liberate iodine. Uranium was found in the iodine deposition zones, although a definite ratio of uranium to iodine was not established. On the basis of these studies, the released species was believed to be a uranium iodide. In contrast to the experiments with helium, elemental iodine, rather than combined iodide, was released from uranium heated to 1 200 °C in air. Uranium iodides are not stable under these conditions, and would not form.

Up to 88% of the iodine released from uranium-aluminum alloy (LITR-type) into air is apparently elemental [62]. In helium, almost all of the iodine is in the reduced form, possibly in combination with or adsorbed on zinc, an impurity within the fuel. No uranium is found in the iodine deposition zones [62]. This would be expected, since the chemical activity of the uranium is greatly lowered by the presence of the aluminum [87].

Chemical state of iodine released from UO_2 and uranium in steam

The release of iodine from uranium metal and UO_2 in steam was studied over the temperature range of 1 100 to 1 300 °C [7, 8]. An average of 80 ± 10% of the iodine released from the uranium metal was collected in the aqueous condensate as iodide ions, about 1% as the iodate and periodate states, and the remainder as elemental iodine. In contrast, the majority of the iodine collected in the UO_2 experiments was elemental, 0.5% was iodate and periodate states, and the remainder was iodide. Although the uranium metal and UO_2 release experiments were similar, the uranium metal reacted with steam to form H_2. The majority of the iodine released from uranium metal undoubtedly reacted with H_2 to form HI; however, the percentage conversion depends on the temperature and H_2 partial pressure.

In UO_2 melting experiments [88] conducted in the containment mock-up facility, iodine released by melting trace-level irradiated UO_2 and allowed to age for 2 h in a stainless steel containment vessel was analysed for the distribution of chemical states by the diffusion-tube method. In these experiments, 75 to 80% of the airborne iodine was apparently in elemental form, 15 to 20% on a small particle about 20 Å in diameter, and about 5% on a particle of 100 Å or greater.

Oxidation states of other fission products released in fuel melts

Oxidation states have been reported for the case where fission products are released from reactor fuels heated to high temperatures in an atmosphere containing some oxygen [89]. Krypton, Xe, I_2, TeO_2+Te, Cs, RuO_4, and Sr are believed to exist in the vaporized state, while the deposited state (assuming an atmosphere containing oxygen and some reducing agent on plate-out surfaces) includes I^-+ I_2, TeO_2, Cs_2O, Ru_2O_3+Ru, and SrO [89] and, in some experiments, MoO_3 [7]. Strontium, Ba, and Zr vaporized in the metallic form from stainless steel clad UO_2 melted in helium. In one experiment, however, they were thought to be present as the oxide [90]. The state was deduced from the results of in-pile fission-product fractionation studies.

Studies directed toward the characterization of released iodine

Deposition of iodine compounds from viscous air flow to diffusion tubes has been employed to determine diffusion coefficients of the compounds [91, 92]. The various species were distinguished chemically by a series of diffusion tubes which have successive surfaces of silver, rubber, and activated carbon. As shown in Fig. 4, the observed diffusion coefficients were 0.08, 0.035, and 0.10 cm²/s, respectively, which correspond respectively to molecular iodine, a compound or group of compounds with a molecular weight of the order of 300 or higher, and a low-molecular-weight compound, such as, for example, HI or CH_3I. This behavior was typical of iodine from two greatly

differing sources. The proportion of iodine present as compounds increased as the I_2 in the air decreased.

In the analysis of iodine distribution from melting trace-irradiated UO_2 in the containment mock-up facility [93], it was found that the passing of air through charcoal (possibly thereby removing organic material) strongly affected the distribution of states of iodine to the extent of changing the amount of organic iodides (charcoal penetrating forms) from 3.7 to 0.4%. Higher concentrations of iodine (fuel with high burn-up) would not be expected to yield so high a conversion to organic forms.

FISSION-PRODUCT TRANSPORT: NATURAL REMOVAL PHENOMENA

Following their release from the fuel, many of the fission products will deposit in close proximity to the fuel and will be of little consequence to the over-all hazard. Others, which may be in the vapor state or attached to particulate matter, will be transported out of the high-temperature melt-down region, where they may condense, agglomerate and settle, or, in the case of volatile vapors and fine particles, may remain airborne for extended periods of time. The ultimate disposition of the emanating fission products depends, to a large extent, on their chemical and physical properties (or those of compounds with which they are associated), as well as those of the particles to which they may become attached.

Deposition

In-pile release studies with stainless steel-clad UO_2 melted in helium showed that in experiments in which nearly all the I, Te, and Cs, and over half the Sr, Zr, Ru, Ba, Ce, and UO_2 were consistently released from the molten fuel, only 80–90% of the released volatiles (i. e., I, Te, and Cs) and less than a few per cent of the nonvolatiles were transported into regions below 1 000 °C [91]. In other experiments with pyrolytic-carbon-coated uranium carbide in air, the in-pile oxidation of the fuel quantitatively released I, Te, Cs, and Ru, and over 80% of these released isotopes were transported out of the high-temperature zone (400 °C), while less than 10% of the Sr and Ba (but not Zr and Ce) got out of this zone.

Other studies have indicated the significance of flow conditions on the distribution of deposited fission products. In one out-of-pile helium loop test at BMI with a heated irradiated UC_2 fuel material as the source of fission products, there was virtually no selective temperature deposition at Reynolds numbers in excess of 5 000 [3, 94, 95], although disturbances in flow patterns inevitably resulted in marked increases in local deposition. Complementary tests have been performed with the in-pile fuel melting facility at ORNL which indicated that the release of the more volatile isotopes (I, Cs, Te) from the high-temperature zone did not appear to be affected by an increase in the helium flow-rate, but the release of the less volatile isotopes from the high-temperature zone increased fourfold with a sixfold increase in flow-rate [90]. Other studies of deposition of selected isotopes as a function of temperature indicate that iodine released from uranium metal into helium deposits on gold, stainless steel, and quartz surfaces at 250 to 350 °C in combined form, probably as a uranium compound. Lanthanum, barium, and caesium deposit on surfaces primarily above 1 000 °C [6, 8, 96]. Several other deposition studies with gross fission products have been reported but do not permit definitive generalizations [1, 38, 96].

A number of mathematical models have been developed in an effort to interpret and predict the results of fission-product deposition. Equations are available for calculating particle deposition in channels and are being used to determine particle sizes [97], but the model is limited to small particles up to 200 Å, laminar flow, isothermal conditions, and walls that serve as a perfect sink for the deposited material. Another model, which assumes that diffusion is the dominating mechanism for deposition, employs the heat-mass analogy to predict fission-product deposition. This has been done with satisfactory results for both laminar and turbulent flow for isothermal conditions [98]. A subsequent adaptation which provides for temperature effects has been employed with some success [99].

A somewhat different model was developed based upon various solutions of Laplace's equation which was a consequence of a simplification of the transport equation assuming Fick's first law [5] .This model has also been adapted to calculate deposition in non-isothermal zones using partial pressures rather than fission-product concentrations in the diffusion equations. The correlations to date with experimental results have also been encouraging.

Agglomeration and settling

Particles tend to coalesce as they contact, resulting in the formation of progressively more coarse particles which eventually settle. Although there is a paucity of experimental information on this effect in relation to particles produced in a reactor accident, conservative calculations [100], however, showed that particles (assumed to be pure iodine) having diameters less than 0.3 μ agglomerate rapidly to 0.3 μ or greater in size, at which point they can be removed by filters. Experimental confirmation of this characterization behavior was subsequently obtained with particles containing fission products released from melted UO_2 [101].

Condensation

Removal of fission products on water condensate has been observed incidental to a number of experiments in which a steam atmosphere was employed [56, 62, 102]. The condensation of steam removed from 40 to 97% [56, 62, 103] of the evolved fission products. Condensation is especially effective when the steam issues under a pool of water as in the pressure-suppression containment design [104, 105].

Retention by coolant

In some reactor systems in which the coolant is presumed to be available as a sink for fission products after their release from the fuel, the retention of fission products may be significant. Experiments have shown that over 99.5% of the gaseous iodine released into a pool of hot sodium will be retained [106–109].

FISSION PRODUCT TRANSPORT: INDUCED REMOVAL PROCESSES

Development studies and tests have been carried out on a number of induced fission-product removal processes, including those making use of particulate filters, adsorbers, scrubbers, foam encapsulation, scavenging, or diffusion boards. The general subject has been reviewed [110], and a series of conferences have been held on the subject of gas cleaning under sponsorship of the USAEC [111–115].

Particulate filters

Radioactive fission products in the form of aerosol particles may be removed from gases by high-efficiency (> 99.95%) particulate filters, which are commercially available. Recommendations have been prepared for the inspection, storage, handling, and installation of such units [116], and methods for testing such filters in place after installation have been developed [117, 118] to ensure their continued effectiveness. Particulate filters in the reactor ventilation systems can be routinely tested [119, 120]. Since an accident involving a reactor that contains appreciable amounts of water would release large volumes of steam and fog, tests have been conducted to determine the durability and efficiency of high-efficiency filters under these conditions. The filters, which were preceded by moisture separators, were undamaged by 10 days of exposure to mixtures of steam and air with entrained water, and they maintained efficiencies of at least 99.5% [9, 121].

The effectiveness of filters and adsorbers has been tested for fission products in the forms released from reactor fuels of various types, melted or otherwise damaged in simulated reactor accidents under various atmospheric conditions, and the results are summarized in Table 7. Filter efficiencies ranging from 40 to 80% for Sr and Ba and 50 to 99% for I, Te, Cs, and Ru were observed for fission products released from metallic uranium fuel elements heated to 1 215 °C in atmospheres of air, steam, or helium [56]. In other tests where UO_2-stainless steel fuel elements clad with

Table 7. Efficiencies of various gas cleaning processes for removing fission products released by destruction of nuclear fuels

Condition and reference	Removal method	Percentage of fission-product removal								U removed (%)
		I	Cs	Te	Ru	Sr	Ba	Zr	Ce	
UO_2 dispersed in stainless steel; melted in air [50]	High-efficiency filter	71	99.7							
Uranium melted in air [56]	Deposition near fuel	10	40	20	85	55	50	40		
	High-efficiency filter	56	93	97	97	53	73			
Uranium melted in steam [56]	Deposition near fuel	60	45	70	60	50	85	50		
	High-efficiency filter	98	55	99	92	80	74			
	Steam condensation	97	80	77	60	40	50			
Uranium melted in helium [56]	Deposition near fuel	90	50	98	60	90	98	40		
	High-efficiency filter	98	49	99	78	38	50			
Aluminum-uranium alloy melted in dry air [71]	Deposition near fuel	2	46							
	High-efficiency filter	46	> 99.9							
	Charcoal + filter	99.97	> 99.99							
Aluminum-uranium alloy melted in moist air [71]	Deposition near fuel	4	33							
	High-efficiency filter	23	> 99.9							
	Charcoal + filter	99.97	> 99.99							
Stainless steel-clad UO_2 melted in air [71]	Deposition near fuel	8	11	20	47					
	High-efficiency filter	30	> 99.9	> 99.9	> 99.9					> 99.9
	Charcoal + filter	98.6	> 99.9	> 99.9	> 99.9					> 99.9
Stainless steel-clad UO_2 melted in-pile in helium [90]	Deposition near fuel	20	41	10	96	99.0	98.9	99.9	99.7	99.99
	High-efficiency filter	92	86		60	48	52	40	52	31

stainless steel were melted in air, the released caesium was removed by an HV-70 filter with 99.7% efficiency, and the released iodine was removed by a CWS-6 type filter with 71% efficiency [50]. In tests in which full-scale metallic fuel elements were melted in air and samples collected at distances up to 5 miles downwind, iodine was the only fission product which penetrated absolute filters to an extent exceeding 5% [122]. When aluminum-uranium alloy fuel elements were melted in dry air and in moist air atmospheres, over 99.5% of the gas-borne caesium was retained by a high-efficiency filter. Iodine was removed by the filter with 46% efficiency in dry air and 23% in moist air [71]. A high-efficiency filter retained $>$ 99.9% of the Cs, Te, Ru, and U, and 30% of the iodine released by melting stainless steel-clad UO_2 in air [71]. In an in-pile melting experiment in which a stainless steel clad UO_2 fuel element was melted in a helium atmosphere, a combination of a roughing filter and an HV-70 filter removed 92% of the iodine, 86% of the caesium, and 42 to 60% of the Ru, Sr, Ba, Zr, and Ce [90]. In general, the less volatile fission products were removed with efficiencies considerably less than might have been anticipated. However the amount of these fission products arriving at the filters was only a very small fraction of the inventory in the fuel because of the low release percentages for these materials.

Adsorbers

Radioactive fission products in gaseous or vapor form may be removed from gases by adsorbers. The removal by adsorption of rare gas fission products and of radioiodine has been reviewed in a series of articles [110, 123, 124]. Activated carbon is the most efficient adsorber for removing radioiodine from gases under ordinary circumstances. Practical adsorbers have been designed having efficiencies greater than 99.9%; efficiencies of 99.99% are commonly observed [27, 51, 125–128]. High efficiencies persist even after extended operation [10, 121, 128] and during operation in a steam atmosphere [10, 121, 129, 130]. Radioiodine has been removed from helium at temperatures up to 700 °C by activated carbon impregnated with metal salts [131, 132]. A combination particulate filter and charcoal adsorber collected 99.95% of the iodine released by melting aluminum-uranium alloy fuels in dry or moist air, and 98.6% of that released by melting stainless steel-clad UO_2 fuel in air [71]. Under certain circumstances iodine can occur in a nonelemental vapor form, especially at very low iodine concentrations [96, 133, 134]. One such iodine vapor species has been observed to penetrate activated charcoal under humid conditions in air but not under dry conditions [96]. Reduced efficiencies for adsorption of radioiodine by charcoal have been observed, especially at very low concentrations, under special circumstances where the iodine had been exposed to impurities from chemical processing plants, metallurgical hot cells, or burning organic materials [27, 128, 135–137]. Methods for testing iodine adsorbers in place after installation have been developed [130] to ensure their continued effectiveness and can be applied routinely [119, 120, 130].

Other absorbents have been investigated to avoid the possible hazards associated with the combustibility of activated charcoal. Silver on various supports removes iodine vapor from air up to 300 °C and from dry steam; however, it is more susceptible than charcoal to interference by impurities or surface contamination [27, 51, 127, 138, 139]. These materials have been placed in iodine removal systems ahead of a charcoal adsorber to reduce the radioiodine decay heat generated in the charcoal. Molecular sieves have been tested for adsorption of radioiodine from gases [27, 126, 140, 141] but also have the disadvantage of moisture interference. Potassium iodide and platinized aluminum have been tested for removing iodine from oxidizing gases at temperatures in excess of 500 °C [71, 142].

Other removal techniques

Other methods have been investigated for removing fission products from gases. Scrubbers can be effective under certain conditions [110]. Scavenging by an aerosol formed in the gas has been tested [12]. Encapsulation in foam appeared attractive on a small scale, but large-scale tests proved disappointing possibly because of entrainment of spray [13, 143]. A diffusion board, which would be porous material surrounding a reactor, proved effective against both iodine and particulates [143].

SUMMARY

From the data now available reasonably accurate estimates of the quantities of fission products released from UO_2 and uranium alloy fuels has been determined under a variety of conditions of temperature, time, atmosphere, and radiation exposure. As far as reactor accident analysis is concerned, the accuracy of such estimates will exceed the accuracy with which the core temperature distribution can be predicted as a function of time, as well as the present understanding of fission-product transport phenomena. Thus recent efforts in the field of fission-product behavior studies have been directed toward a better understanding of transport phenomena including both natural and induced removal processes. This, in turn, has necessitated the determination of the physical and chemical states of the released fission products (characterization) in order to understand the transport phenomena. Laboratory tests have already identified various forms of different fission-product elements, particularly iodine,

and have indicated, if incompletely, their behavior under a variety of conditions simulating those which might exist in a reactor accident. Particulate filters and iodine adsorbers, generally capable of removal efficiencies in excess of 99 %, have under special circumstances proved to be much less effective.

The behavior of released fission products is, however, expected to be a function of the geometry and environment (including materials and their surface conditions) in the containment system, as well as the nature of the released fission products themselves. In order to verify that the behavior of released fission products in actual containment systems may be extrapolated from existing laboratory-scale tests, three facilities, each permitting a closer approximate to a full-size reactor melt-down, have been, or will be, built. These facilities are the nuclear safety pilot plant (NSPP) at ORNL, the containment systems experiment (CSE) at Hanford, and the loss of flow test (LOFT) at NRTS. Of these three facilities, only the NSPP has been constructed; it was placed in operation early this year. If the fission-product transport behavior in these facilities substantiates that which may be extrapolated from the laboratory tests now well under way, a much more realistic evaluation of the consequence of reactor accidents will become available. The experimental program may be expected to demonstrate that the fission-product inventory available for leakage out of a containment system may be orders of magnitude less than that which has hitherto been postulated [144, 145], and this would be consistent with interpolations from the limited data now available from actual reactor accidents [146]. In the United States, at least, such conclusions could not fail to have a profound effect on reactor siting criteria [144] and containment requirements [147].

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/285 Etats-Unis d'Amérique

Expérience acquise aux Etats-Unis concernant la libération et le transport de produits de fission à l'intérieur des enveloppes de sécurité dans des accidents de réacteurs simulés

par W. B. Cottrell *et al.*

A la suite des études générales entreprises dans le cadre du programme de sécurité des réacteurs de la Commission de l'énergie atomique des Etats-Unis ainsi que des recherches plus particulières conduites au titre de divers programmes de réacteurs, plus d'une douzaine de laboratoires des Etats-Unis rassemblent des renseignements sur les produits de fission libérés par divers combustibles dans des conditions d'accidents simulés ainsi que sur le transport et l'élimination de ces produits de fission.

Si jusqu'à présent la plupart des travaux ont porté sur les produits de fission libérés par les combustibles à base d'uranium métallique, d'alliages d'uranium et d'oxyde uraneux, on dispose cependant de quelques renseignements sur les produits de fission libérés par les combustibles UC et BeO/UO_2. Pour tous les combustibles, les renseignements sont donnés en fonction de plusieurs paramètres, qui comprennent généralement la température (jusqu'à la fusion), le temps (plusieurs heures), le taux de combustion (jusqu'à 11 000 MWj/t) et l'atmosphère ambiante (vapeur, air, He, CO_2). On a obtenu la plupart des renseignements hors pile en utilisant des méthodes d'induction, de résistance ou de chauffage du plasma; quelques résultats ont également été obtenus au moyen d'expériences en pile.

En règle générale, lorsque UO_2 est fondu on peut s'attendre à la libération de grandes quantités de gaz rares et d'iode, qui s'accompagne fréquemment de la libération de quantités importantes (pouvant aller jusqu'à 100%) de Cs et de Te. Des quantités comparables sont libérées à des températures un peu plus basses, aux environs de 1 090 °C, mais cette libération dure alors plusieurs heures. D'autre part, Sr libéré par UO_2 était très inférieur à 1% sauf à des températures supérieures à 1 600 °C dans l'hélium. Dans le cas de l'oxyde comme du métal, la combustion a accéléré la libération. Toutefois, l'influence de la combustion sur la libération fractionnée des produits de fission s'échappant du métal est jugée peu importante par rapport à celle d'autres variables. Avec l'uranium métallique, on obtient une libération fractionnée similaire à celle qui se produit avec UO_2. Les principaux sujets des travaux expérimentaux sont, outre la mesure des quantités libérées, la détermination de la forme physique et chimique des produits de fission libérés, leur transport (principalement le dépôt) et leur séparation de l'air. Toutefois, les nombreux résultats signalés au sujet du dépôt ne se prêtent qu'à un petit nombre de généralisations qualitatives — par exemple, les produits non volatils se déposent à proximité du point de libération à des températures supérieures à 1 000 °C.

Dans les études sur la détermination de la forme des produits de fission, on s'est occupé surtout de l'iode en raison des risques potentiels relativement plus grands qu'il présente; ces études ont abouti à des résultats intéressants sinon absolument concluants. Dans l'hélium à 1 200 °C l'iode se dégage du métal irradié sous une forme combinée, qui se dissocie partiellement à des températures plus élevées et entièrement sous l'effet d'une oxydation ultérieure pour donner de l'iode pur.

Les études sur la séparation des produits de fission, notamment par pulvérisation, lavage, balayage et filtration ont été assez poussées et, pour ce qui est de la filtration, ont donné des résultats concluants. Les absorbeurs au charbon de bois pour l'iode et les filtres pour les particules se sont toujours révélés efficaces, le pouvoir de séparation dépassant 99%, même dans une atmosphère de vapeur.

Les recherches actuelles sont complétées par plusieurs expériences à l'échelle industrielle qui ont pour objet de montrer dans quelles limites ont peut extrapoler les données obtenues dans les laboratoires et se servir des données ainsi extrapolées pour évaluer le comportement des produits de fission en cas d'accident de réacteur.

A/285 США

Опыт США по выбросу и переносу продуктов деления внутри защитной системы в условиях имитированной аварии реактора

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Данные по утечке и удалению продуктов деления, выделяемых из различных топлив при имитировании условий аварии реактора, получают более чем в двенадцати различных лабораториях США в результате общих исследований, проводимых по программе безопасности реакторов, утвержденой КАЭ США. Кроме того, ведутся более специализированные исследования по другим реакторным программам.

До настоящего времени большая часть работ по выбросу продуктов деления касалась металлического урана, урановых сплавов и топливных материалов из двуокиси урана, хотя имеются некоторые данные по выходу продуктов деления из топлива в виде UC и BeO/UO_2. Данные по всем видам топлива даются в виде зависимости ряда параметров, таких, как температура (до точки плавления), время (несколько часов), выгорание (до 11 000 *Мвт·сутки/т*) и окружающая атмосфера (пар, воздух, гелий, двуокись углерода). Большинство данных было получено вне реактора при помощи индуктивных методов, сопротивления или плазменной горелки, хотя некоторые данные получены внутри реактора.

Как правило, выброса инертных газов и иода следует ожидать при расплавлении двуокиси урана часто одновременно со значительной (до 100%) утечкой цезия и теллура. Сравнимые утечки были получены при более низких температурах ~1090° С, но для этого потребовалось несколько часов. С другой стороны, выход стронция из двуокиси урана составлял значительно меньше 1%, за исключением утечек в гелии при высоких температурах >1600° С. Как в окисле, так и в металле выброс активных продуктов с выгоранием увеличивался. Однако вли-

яние выгорания на относительную утечку продуктов деления из металла сравнительно мало по сравнению с другими факторами. Относительные утечки металлического урана совпадали с утечками для UO_2. Кроме измерения утечек и выяснения характеристик радиоактивности, попавшей в окружающую среду (с точки зрения ее физических и химических свойств), ее транспортировка и главным образом осаждение и удаление ее из воздуха составили другие направления экспериментальной работы. Однако многие отчеты по осаждению радиоактивности дают только ряд качественных обобщений, как, например, нелетучие продукты деления осаждаются вблизи от места утечки при температурах выше 1000° С.

Изучение поведения иода, связанное с его относительной опасностью, дало интересные, хотя и не вполне окончательные результаты. В гелии при температуре 1200° С иод выделялся из облученного металла в комбинированном виде и частично диссоциировался при высоких температурах и при последующем окислении превращался в элементарный иод.

Изучение методов удаления продуктов деления, включая опрыскивание, скрубберы, очистку и фильтрацию, было проведено в широких масштабах и дало окончательные результаты в отношении фильтров. Надежная работа адсорбентов из древесного угля для иода и фильтров для частиц была многократно продемонстрирована с эффективностью очистки выше 99% даже в атмосфере пара.

Данные исследования продолжаются и дополняются результатами опытов более крупного масштаба, такими, как опытная установка по ядерной безопасности (NSPP), опыт по снижению расхода теплоносителя (LOFT) и эксперимент по исследованию систем удержания (CSE), целью которых является оценка возможности экстраполяции лабораторных данных для оценки поведения радиоактивности в условиях аварии реактора.

A/285 Estados Unidos de América

Experiencia estadounidense en el desprendimiento y transporte de productos de fisión en los sistemas de contención en condiciones simuladas de accidente en un reactor

por W. B. Cottrell *et al.*

Como consecuencia de los estudios generales, patrocinados por la USAEC dentro del Programa sobre Seguridad de los Reactores Nucleares, así como de investigaciones más concretas en relación con diversos programas de reactores, en más de una docena de laboratorios diferentes de los Estados Unidos se están obteniendo datos sobre desprendimiento, transporte y separación de los productos de fisión que escapan de diversos combustibles en condiciones simuladas de accidente.

Hasta la fecha la mayor parte de los trabajos realizados sobre desprendimiento de productos de fisión lo han sido sobre materiales combustibles de uranio metálico, aleaciones de uranio o UO_2, aunque existen también datos limitados sobre el mismo tema en combustibles de UC y BeO/UO_2. En la memoria se exponen los datos para todos los combustibles en función de un número de parámetros que incluyen generalmente la temperatura (hasta la fusión), el tiempo (varias horas), grado de quemado (hasta 11 000 MWd/t) y atmósfera ambiente (vapor, aire, He, CO_2). La mayor parte de los datos se obtuvieron en experiencias fuera del reactor utilizando técnicas de calentamiento por inducción, resistencia o calentamiento de plasma; se incluyen algunos resultados obtenidos dentro del reactor.

En general, cuando se funde el UO_2 se puede contar con un desprendimiento cuantitativo de los gases nobles y el yodo, acompañado de un desprendimiento también importante (hasta del 100%) de Cs y Te. Se obtienen unos resultados semejantes a temperaturas algo más bajas, unos 2 000 °F (1 090 °C), aunque para ello se requieren varias horas. Por otra parte, el Sr desprendido del UO_2 era considerablemente inferior al 1%, excepto a temperaturas elevadas, superiores a 2 900 °F (1 600 °C) y en atmósfera de helio. Tanto para el óxido como para el metal, el grado de quemado acelera la velocidad de desprendimiento. Sin embargo, la influencia del grado de quemado en la fracción desprendida de los productos de fisión, en el caso de combustible metálico, se considera de importancia secundaria respecto a las otras variables. Para el uranio metálico se obtienen fracciones de desprendimiento semejantes a las que resultan para el UO_2. Además de las medidas de la cantidad desprendida, la identificación de la actividad liberada (sus formas física y química), su transporte (principalmente la deposición) y su separación del aire constituyen las restantes cuestiones importantes de trabajo experimental. Sin embargo, la variación en los resultados obtenidos sobre deposición se presta únicamente a unas cuantas generalizaciones cualitativas como, por ejemplo, la condensación de los no volátiles junto al punto de desprendimiento a temperaturas superiores a 1 830 °F (1 000 °C).

Los estudios de identificación han versado principalmente sobre el comportamiento del yodo a causa del riesgo potencial que supone y han dado resultados interesantes aunque no completamente definitivos. En helio a 2 200 °F (1 200 °C) el yodo se desprende del metal irradiado en forma combinada, que está parcialmente disociada a temperaturas superiores, y aparece posteriormente como yodo elemental cuando se oxida.

Los estudios sobre separación de productos de fisión, incluyendo rociados, lavados, depuración han sido muy amplios y, en el caso de los filtros, definitivos. Se

ha demostrado concluyentemente el buen funcionamiento de los adsorbentes de carbón para el yodo y de los filtros para las partículas, llegándose a eficacias de separación superiores al 99%, incluso en atmósfera de vapor.

En general, las investigaciones actuales continúan y se suplementan con experimentos a mayor escala como, por ejemplo, la planta piloto de seguridad nuclear (NSPP), el ensayo de pérdida de flujo (LOFT) y el experimento sobre sistemas de contención (CSE), que se realizan para demostrar la validez o limitación de las extrapolaciones de datos de laboratorio al aplicarlos a la evaluación del comportamiento de la actividad en los accidentes de los reactores.

Evaluation of post-incident cooling systems of light-water power reactors

By H. Uchida, A. Oyama and Y. Togo *

The purpose of this paper is to evaluate the effectiveness and importance of post-incident cooling systems of light-water power reactors, both theoretically and experimentally. Theoretical evaluations are made based on JPDR (Japan power demonstration reactor, a 12.5 MW(e) BWR) design data. Experimental evaluations are being conducted in project SAFE (Safe assessment and facilities establishment project), which has been established by a specialist committee of the Japan Atomic Industrial Forum for the study of nuclear safety. Specialists in various fields (universities, research laboratories and manufacturers) attend this committee.

THEORETICAL EVALUATION

Evaluation of the core spray system

A core spray system is installed in some light-water power reactors to prevent substantial fission product (F.P.) release, in case of a loss-of-coolant accident. This system is designed to prevent any fuel perforation or melting; however, non-uniformity of the spray flow across the core might allow some perforation or melting. This process is evaluated in detail and amounts of F.P. release are calculated.

A sketch of the JPDR core spray is shown in Fig. 1. Calculations are made following the chart shown in Fig. 2.

Calculation of f_{sm} and Ψ_s (see Table 1)

Based on R-Z flux distribution [1], segment peaking factors are obtained for each fuel segment. In the JPDR core (Fig. 3), there are 72 fuel assemblies each containing 36 fuel rods, while each rod is composed of two (upper and lower) fuel segments. Therefore, there are 5 184 fuel segments. For the purpose of calculation, 72 upper and lower fuel assemblies are classified into nine groups (i.e., A_1, A_2, ..., A_9 and B_1, B_2, ..., B_9) respectively, according to distances from the centre of the core. 5 184 segments are grouped into A_{ij} and B_{ij} ($i = 1, 2, \ldots, 9$ and $j = 1, 2, 3, 4$) according to segment peaking factors.

Table 1. List of symbols used

A_i, B_i	:	Upper and lower fuel bundle
A_{ij}, B_{ij}	:	Upper and lower fuel segment
F_B	:	Fuel bundle spray flow distribution factor
F_s	:	Fuel segment spray flow distribution factor
F_{sd}	:	Maximum value of F to give clad rupture
(ΣN_τ)	:	F.P. gas production
$(\Sigma N_\tau)_p$	:	F.P. gas stored in fuel plenum
P_{in}	:	Fuel segment inner pressure
P_d	:	Inner pressure to give clad rupture
T_c	:	Fuel clad temperature
T_{c_2}	:	Fuel clad outer surface temperature
T_{c_5}	:	Core spray temperature
T_d	:	Temperature to give clad rupture
W_B	:	Fuel bundle spray flow
W_{cs}	:	Core spray flow
f_B	:	Fuel bundle peaking factor
f_s	:	Fuel segment peaking factor
f_{sm}	:	Maximum value of f for each segment
f_l	:	Local peaking factor across fuel bundle
K	:	Rate of F.P. gas diffused to plenum
Φ'	:	Decay neutron flux heat generation
Ψ_s	:	Fuel segment burn-up peaking factor

Based on R-Z flux distribution, segment peaking factors and segment burn-up peaking factors Ψ_s are calculated. Some examples are given in Table 2 and Table 3.

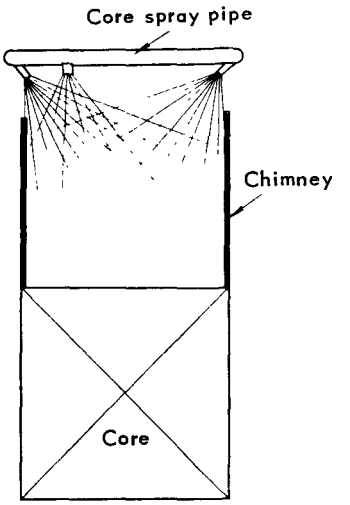


Figure 1. Sketch of core spray

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Calculation of segment clad rupture temperature, T_d

Total F.P. gas production $\Sigma N\tau$ is calculated as a function of f_{sm}. The rate of F.P. gas diffused to a fuel plenum, K, is shown in Fig. 4 which is based on experimental data. $\Sigma N\tau$ multiplied by K gives $(\Sigma N\tau)_p$. Taking account of residual gases in a fuel segment (helium and volatile material in a UO_2 pellet) the segment inner pressure is calculated as shown in Fig. 5. Intersections of curves of P_{in} and P_d give the clad rupture temperature T_d, which is shown in Fig. 6 as a function of f_{sm}.

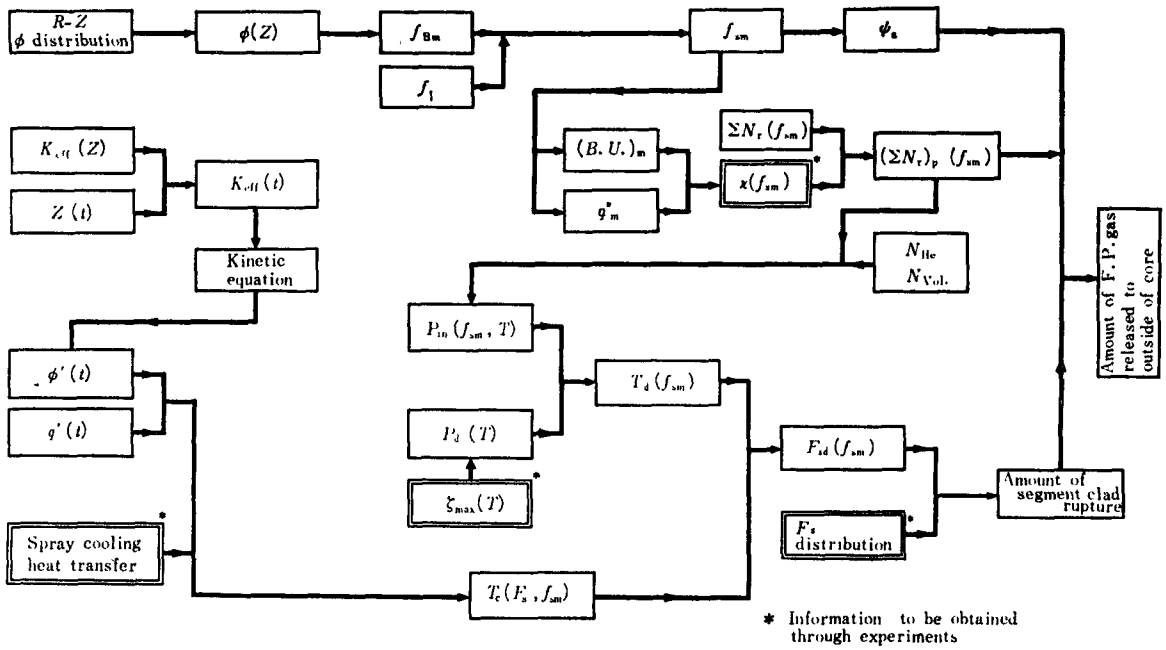
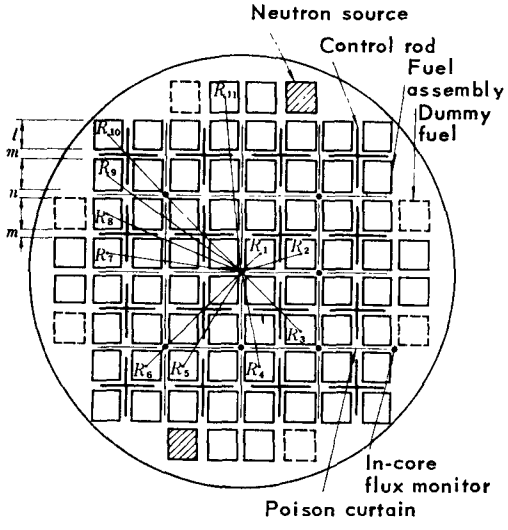


Figure 2. Calculation chart of released F.P. for the maximum credible accident



$R_1=9.35$ cm	$R_8=52.4$
$R_2=21.8$	$R_9=58.7$
$R_3=29.3$	$R_{10}=68.0$
$R_4=34.5$	$R_{11}=61.6$
$R_5=39.7$	$l=12.21$ cm (4.807″)
$R_6=48.0$	$m=1.903$ cm (0.749″)
$R_7=48.4$	$n=1.003$ cm (0.395″)

Figure 3. Arrangement of JPDR core

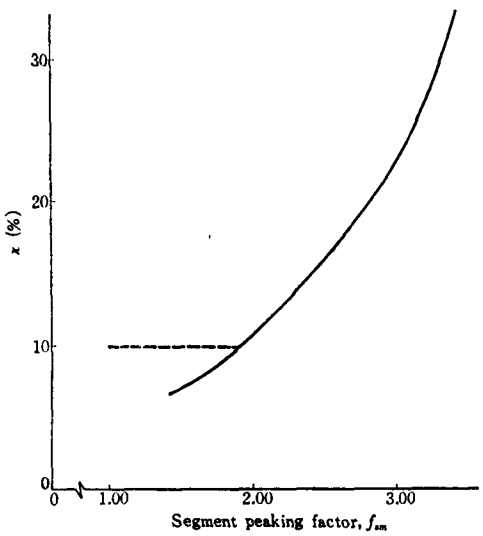


Figure 4. F.P. gas release rate to segment plenum, κ.

Values of κ less than 10% are assumed to be 10% for conservatism

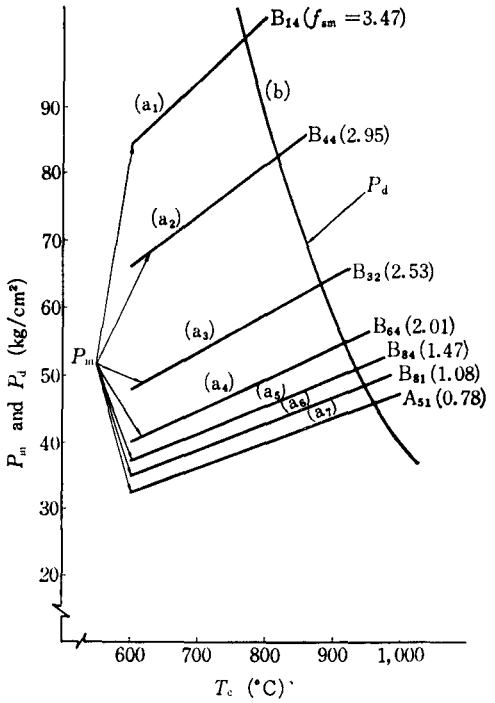


Figure 5. Calculation of segment clad rupture pressure and temperature

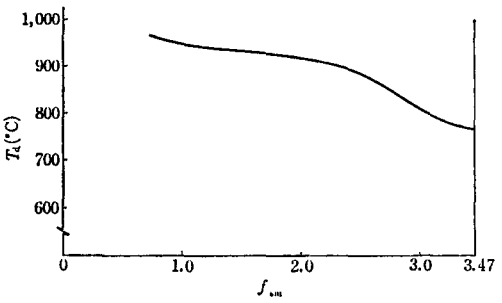


Figure 6. Segment clad rupture temperature, T_d

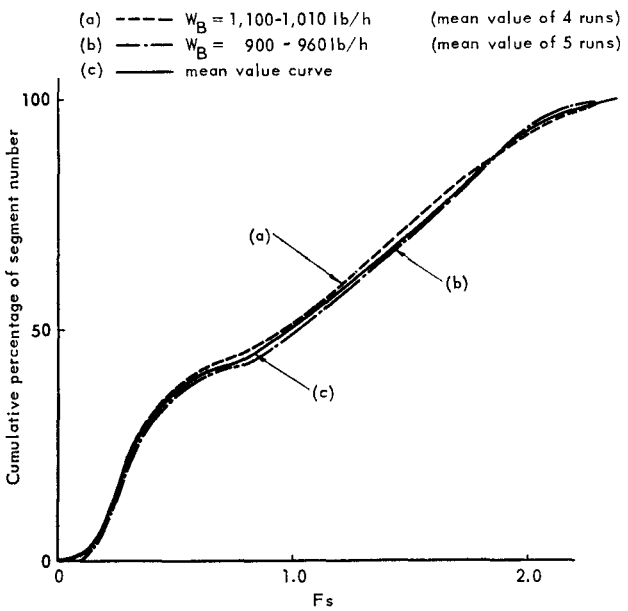


Figure 7. Cumulative percentage of segment flow rate distribution

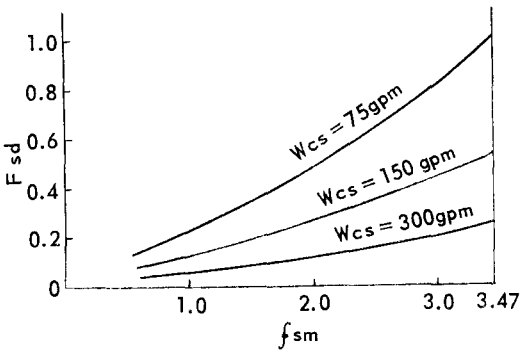


Figure 8. Spray flow rate to give clad rupture, F_{sd}

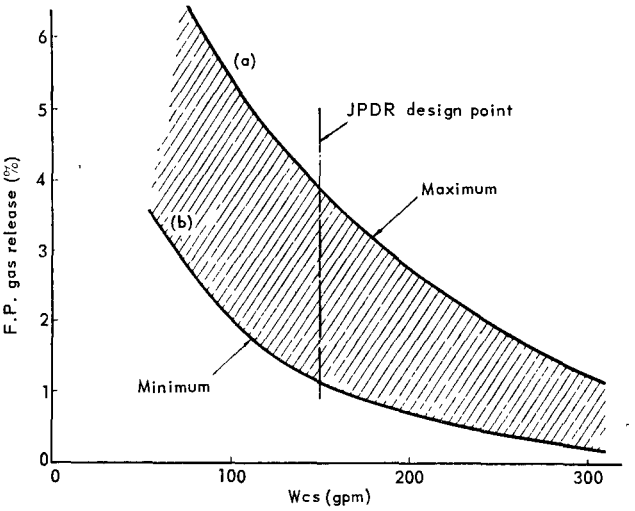


Figure 9. F.P. gas release versus spray flow rate

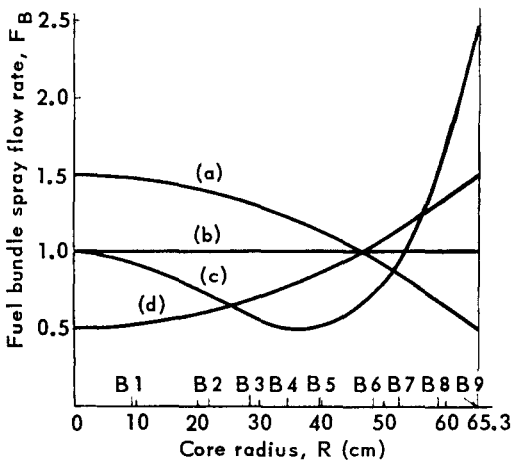


Figure 10. F_B distribution model

Table 2. Segment peaking factor, f_{sm}

Fuel bundle No.	fB_m	Number	Segment No.	f_1	f_{sm}	Number
A_1	1.07	4	A_{11}	0.91	0.98	64
			A_{12}	1.01	1.08	24
			A_{13}	1.13	1.21	44
			A_{14}	1.24	1.33	12
A_2	1.04	8	A_{21}	0.91	0.95	128
			A_{22}	1.01	1.05	48
			A_{23}	1.13	1.18	88
			A_{24}	1.24	1.29	24
B_1	2.80	4	B_{11}	0.91	2.55	64
			B_{12}	1.01	2.83	24
			B_{13}	1.13	3.17	44
			B_{14}	1.24	3.47	12
B_2	2.60	8	B_{21}	0.91	2.37	128
			B_{22}	1.01	2.63	48
			B_{23}	1.13	2.94	88
			B_{24}	1.24	3.23	24

Table 3. Segment burn-up peaking factor, Ψ_s

Fuel bundle No.	Segment No.	ψ_s
A_1	A_{11}	0.68
	A_{12}	0.76
	A_{13}	0.85
	A_{14}	0.93
A_2	A_{21}	0.66
	A_{22}	0.74
	A_{23}	0.83
	A_{24}	0.91
B_1	B_{11}	2.03
	B_{12}	2.26
	B_{13}	2.53
	B_{14}	2.77
B_2	B_{21}	1.93
	B_{22}	2.14
	B_{23}	2 39
	B_{24}	2.63

Calculation of maximum and minimum F.P. gas release

Based on the experimental data of spray cooling heat transfer, the clad temperature T_c is calculated as a function of F_s and f_{sm}. Comparing T_c with T_d, the maximum segment spray flow rate to give a clad rupture, F_{sd}, is calculated as in Table 4. The segment spray flow distribution is also experimentally given in Fig. 7.

In Table 5 the maximum F.P. gas release is calculated for the spray flow rate of 150 gpm which is the JPDR design figure. In making the calculation, assumptions are made that the segment spray flow distribution is unchanged as shown in Fig. 7 and that this is random phenomena. In Table 5, B_{14} is only 8.3% of the total segments. The cumulative sum from the top line in column 8 must not exceed the percentage shown in column 7; for instance, for the fuel bundle B, the amount of ruptured segments is 8.3% for B_{14} (B_{14} is ruptured entirely), 28.7% for B_{13} and none for B_{12} and B_{11}. It is unreasonable to assume that more than 28.7% of B_{13} is ruptured because ruptured segments of B_{14} plus B_{13} exceeds 37% (the cumulative sum of F_s). It is self-explanatory that a combination such as to rupture segments with larger f_{sm} first, gives the maximum F.P. gas release. In the same manner, the minimum F.P. gas release is calculated for the core spray flow of 150 gpm. The above calculations are repeated for the core spray flows of 75 gpm and 300 gpm. Results are shown in Figs. 8 and 9.

Evaluation of nonuniformity of the fuel bundle spray flow distribution

To evaluate the effects of nonuniformity of F_B distribution on the F.P. gas release, F_B distribution models as shown in Fig. 10 are arbitrarily adopted. From Table 5, the maximum F.P. gas release for each fuel bundle is calculated. Results are given in Fig. 11, together with the minimum F.P. gas release. Calculated total F.P. maximum gas releases are, 3.29, 3.86, 4.75 and 4.44% for models (*a*), (*b*), (*c*) and (*d*) in Fig. 10. Minimum values are 0.82, 1.11, 1.92 and 1.59%, respectively. These results show that effects of nonuniformity of F_B distribution on the F.B. gas release is of secondary importance.

Temperature transient analysis

One example of fuel segment temperature transients is shown in Fig. 12.

Stability criteria [2]

Criteria for stable spray cooling are obtained with the assumption that all spray drops are spherical. Results are given in Table 6.

Evaluation of a container spray system [2]

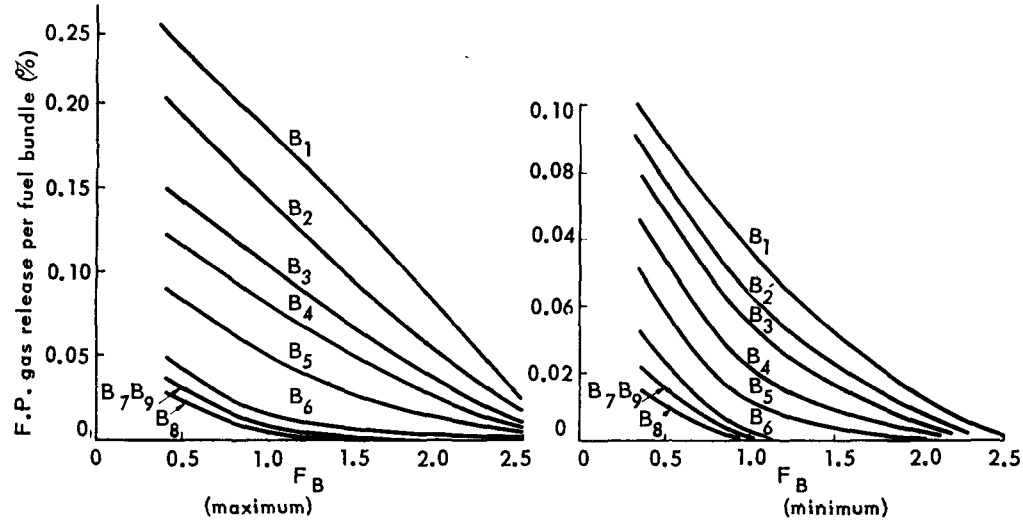
In evaluating a container spray system, the reduction factors of F.P. gas releases are calculated, and the relative significance of a container spray system is discussed. Results are shown in Table 7.

Table 4. Spray flow rate to give clad rupture, F_{sd}

f_{sm}	T_d	$(T_c-T_{es})^{3/4}$	q''	γ	W_B	F_{sd}
	°C		cal/cm²s	$\times 10^{-3}$	g/s	
3.47	767	131	1.06	8.08	60.0	0.50
3.00	810	137	0.915	6.67	48.6	0.41
2.50	885	148	0.762	5.15	37.4	0.31
2.00	917	152	0.610	4.02	28.6	0.24
1.50	933	155	0.458	2.96	21.0	0.18
1.00	948	157	0.305	1.94	13.5	0.11
0.75	963	159	0.229	1.44	10.0	0.08

Table 5. Calculation of maximum F.P. gas release

Fuel bundle No.	Number of fuel bundles %	Segment No.	f_{sm}	Number of segments %	F_{sd}	Cumulative sum of F_s %	Amount of ruptured segments %	Core %	$\varkappa$ %	ψ_s	F.P. gas release %
A_1	2.78	A_{14}	1.33	8.3	< 0.15	3	3.0	0.08	10	0.93	0.008
		A_{13}	1.21	30.5	0.13		0	0			0
		A_{12}	1.08	16.7			0	0			0
		A_{11}	0.98	44.5			0	0			0
A_2	5.55	A_{24}	1.29	8.3	0.14	3	3.0	0.17	10	0.91	0.015
		A_{23}	1.18								
		A_{22}									
		A_{21}									
A_3	2.78	A_{34}	1.27	8.3	0.13	3	3.0	0.09	10	0.89	0.008
		A_{33}	1.15								
A_4	5.55	A_{44}	1.20	8.3	0.13	3	3.0	0.17	10	0.86	0.014
		A_{43}									
A_5	5.55	A_{54}	1.06	8.3	0.11	2	2.0	0.11	10	0.82	0.009
		A_{53}									
A_6	8.35	A_{64}	0.86	8.3	0.09	1	1.0	0.08	10	0.67	0.005
		A_{63}									
A_7	5.55	A_{74}	0.76	8.3	0.09	1	1.0	0.06	10	0.58	0.004
		A_{73}									
A_8	5.55	A_{84}	0.76	8.3	0.09	1	1.0	0.06	10	0.50	0.004
		A_{83}									
A_9	8.35	A_{94}	1.07	8.3	0.11	2	2.0	0.17	10	0.72	0.012
		A_{93}									
										Sub-total =	0.08%
B_1	2.78	B_{14}	3.47	8.3	< 0.50	37	8.3	0.23	34	2.77	0.22
		B_{13}	3.17	30.5	0.43	32	28.7	0.80	26	2.53	0.53
		B_{12}	2.83	16.7	0.37	28	0	0			0
		B_{11}	2.55	44.5	0.32	24	0	0			0
B_2	5.55	B_{24}	3.23	8.3	0.44	33	8.3	0.46	28	2.63	0.34
		B_{23}	2.94	30.5	0.39	29	24.7	1.37	25	2.39	0.82
		B_{22}	2.63	16.7	0.33						
		B_{21}	2.37	44.5							
B_3	2.78	B_{34}	3.10	8.3	0.42	31	8.3	0.23	23	2.48	0.13
		B_{33}	2.83	30.5	0.37	28	22.7	0.63	20	2.26	0.29
		B_{32}	2.53	16.7							
B_4	5.55	B_{44}	2.95	8.3	0.39	29	8.3	0.46	21	2.33	0.23
		B_{43}	2.69	30.5	0.34	25	20.7	1.15	17	2.13	0.42
		B_{42}	2.40	16.7							
B_5	5.55	B_{54}	2.59	8.3	0.32	24	8.3	0.46	17	2.15	0.17
		B_{53}	2.36	30.5	0.29	21	15.7	0.87	14	1.95	0.24
		B_{52}	2.11	16.7							
B_6	8.35	B_{64}	2.01	8.3	0.24	12	8.3	0.69	11	1.68	0.13
		B_{63}	1.83	30.5	0.21	9	3.7	0.31	10	1.54	0.05
		B_{62}	1.64	16.7							
B_7	5.55	B_{74}	1.74	8.3	0.20	8	8.0	0.44	10	1.44	0.06
		B_{73}	1.58	30.5							
B_8	5.55	B_{84}	1.47	8.3	0.16	5	5.0	0.28	10	1.36	0.04
		B_{83}	1.34								
B_9	8.35	B_{94}	1.74	8.3	0.20	8	8.0	0.67	10	1.67	0.11
		B_{93}	1.58								
										Sub-total =	3.78%
										Total =	3.86%



Fuel Bundle No.	Number
B_1	4
B_2	8
B_3	4
B_4	8
B_5	8
B_6	12
B_7	8
B_8	8
B_9	12

Figure 11. Maximum and minimum F.P. gas release per fuel bundle

EXPERIMENTAL EVALUATION

In order to evaluate experimentally the theoretical considerations mentioned above, the following research programmes have been established:

(*a*) Investigations of transient phenomena in the reactor in the case of loss-of-coolant by a rupture of the primary cooling system;

Table 6. Stability criteria of core spray flow

Fuel rod	$\psi_{sA}+\psi_{sB}$	v_{up} cm/s	Stability criteria cm	Remarks
$A_{14}+B_{14}$	3.70	353	$r_S > 2.05 \times 10^{-2}$	Max heat generation
$A_{12}+B_{12}$	3.02	288	$r_S > 1.86 \times 10^{-2}$	
$A_{31}+B_{31}$	2.47	237	$r_S > 1.68 \times 10^{-2}$	
$A_{62}+B_{62}$	1.92	183	$r_S > 1.47 \times 10^{-2}$	
$A_{81}+B_{81}$	1.36	130	$r_S > 1.24 \times 10^{-2}$	Min heat generation

(*b*) Studies on the effectiveness of the core spray in preventing the temperature rise and melting of the fuel elements;

(*c*) Studies on the effectiveness of the container spray in the suppression of pressure rise in the container vessel and its wash-out effect on the radioactive gas and vapour in the container;

(*d*) Studies on the leakage of gas to the outside of the container at the penetration of pipes or cables on the container wall.

Experimental research on the above-mentioned post-incident cooling systems is being carried out.

Transient phenomena following pipe rupture in the primary cooling system

In order to investigate the transient phenomena in the case of rupture in the primary cooling system, experiments conducted at Hitachi Ltd. were as described below.

Table 7. F.P. release reduction factor

	^{131}I	^{133}I	^{135}I	Sr	Solid F.P.	Rare gas
Core spray F_1	20	20	20	2 000	2 000	20
(Prevention of UO_2 melt-down) F_{11}	(8)	(8)	(8)	(800)	(800)	(8)
(Prevention of clad perforation) F_{12}	(2.5)	(2.5)	(2.5)	(2.5)	(2.5)	(2.5)
Enclosure F_2	6.8×10^3	6.9×10^3	7.1×10^3	11.5×10^3	—	2.5×10^3
(Shell effect) F_{21}	(180)	(180)	(180)	(180)	(180)	(180)
(Internal pressure reduction due to enclosure spray) F_{22}	(3.4)	(3.4)	(3.4)	(3.4)	(3.4)	(3.4)
(Natural decay) F_{23} . . .	(1.05)	(1.5)	(2.5)	(1.0)	—	(4.2)
(Wash-out effect) F_{24} . .	(10.7)	(7.6)	(4.7)	(19)	—	(1.0)
Total reduction factor F . .	1.4×10^5	1.4×10^5	1.4×10^5	2.3×10^7	—	5.0×10^4

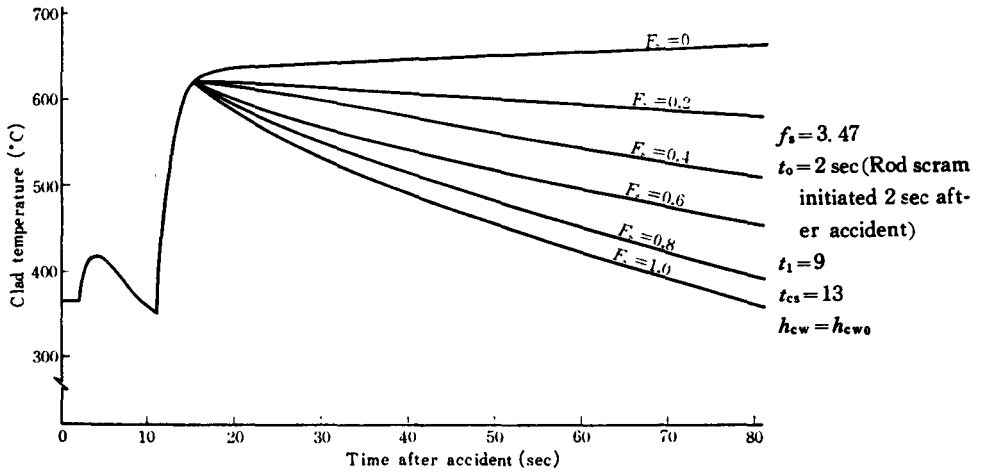


Figure 12. Effect of core spray flow on clad temperature

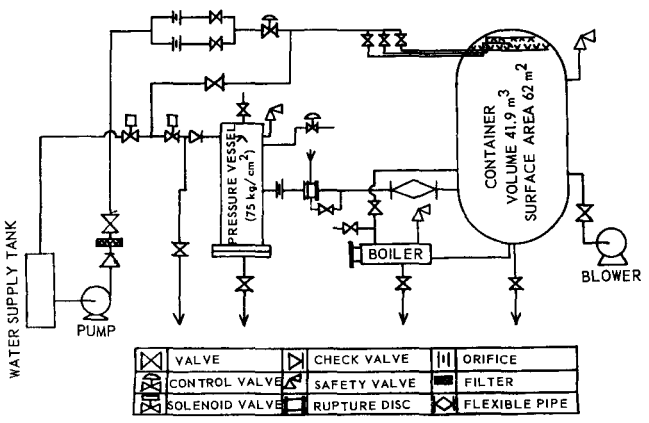


Figure 13. Apparatus for rupture test

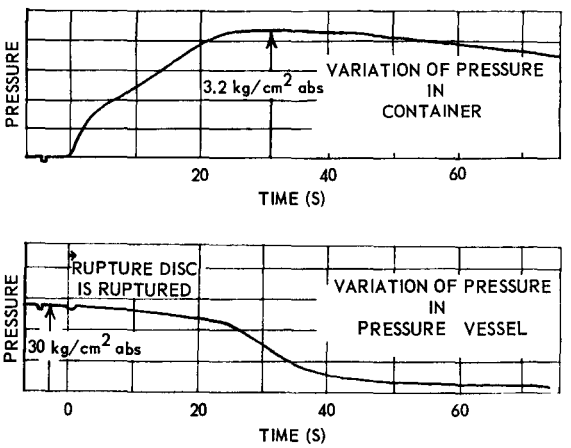


Figure 14. Pressure variation in container and pressure vessel in rupture test

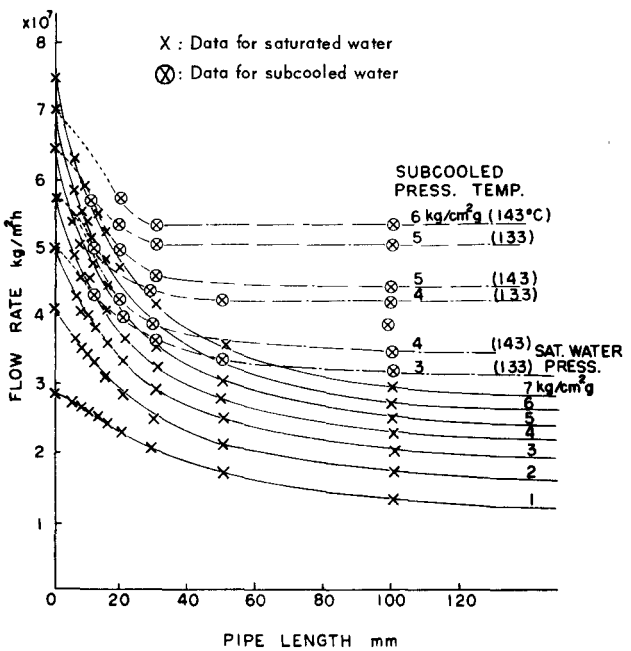


Figure 15. Discharge from short pipe

Figure 13 shows the experimental apparatus. The rupture disc which was installed in the pipe (130 mm id) between the model pressure vessel and the model container was ruptured, and high pressure and high temperature water was discharged into the container. In the experiment, the transient phenomena in the pressure vessel were observed and the pressure and temperature changes in the pressure vessel and the container were recorded. As an example, Fig. 14 shows the results of the experiment when the initial quantity of water in the pressure vessel was 0.6 ton, the initial temperature 234 °C, the initial pressure 30 kg/cm² abs and a 30 mm diameter orifice was set in the pipe.

The basic study of the discharge of high pressure and high temperature water has been done by Uchida *et al.* at the University of Tokyo. In discharging the saturated water or the subcooled water with pressures up to 7 kg/cm²g to the atmosphere through the pipe (4 mm id), the effect of the pipe length (from 0 to 2.5 m) on the discharge has been obtained. The results are as follows: when the pipe length is as short as several mm, which corresponds to the knife-edged orifice, both fluids are

discharged in a supersaturated condition, and the discharge coefficient is the same as in one-phase flow. For an increase in pipe length in the range of tens of mm the discharge decreases steeply, but for a pipe length above 100 mm the decrease of discharge does not depend on the pipe length so much. In the long pipe we could confirm the existence of the critical quantity of flow which depended not on the back pressure but only on the initial pressure in the pressure vessel. Some of these results are shown in Figs. 15 and 16. In Fig. 16, results about critical flow are compared with the well-known Linning and Fauske's theoretical predictions. The experimental results obtained by the authors are larger than the theoretical values of Linning and Fauske.

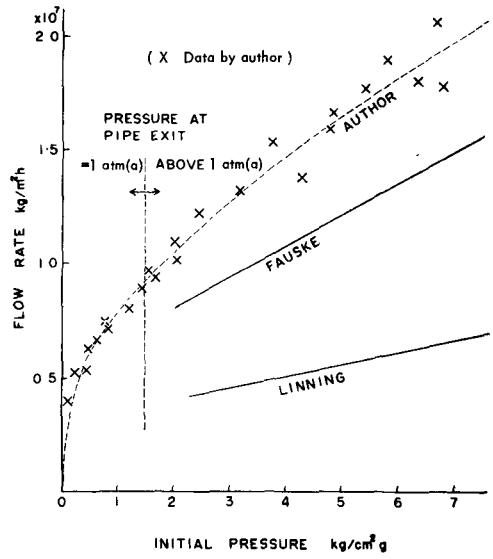


Figure 16. Comparison of discharge from long pipe (4 mm id, length 2 500 mm) to Linning and Fauske's theoretical prediction

Effectiveness of the core spray system

The experimental study of the core spray, required for spraying pressurized water on to the fuel elements to prevent the steep rise of temperature and melting, has been conducted at Hitachi Ltd. and NAIG (Nippon Atomic Industry Group Co.). The following results on the effects of arrangement and dimensions of the nozzles on the spray distribution were obtained.

(*a*) In the case of the single nozzle test, water distribution is fairly uniform, but when three nozzles are used interference between their sprays arises, causing water distribution to be slightly nonuniform.

(*b*) In the case of the fuel assembly model test, water distribution with a three-nozzle spray is more uniform than a single-nozzle spray. This is because the device provided on the upper structure of the fuel assembly model acts as an obstacle making the water distribution with the three-nozzle spray more uniform. The experiment showed that the ratio of minimum-to-average flow per rod was 0.3–0.4 for the six-row, six-

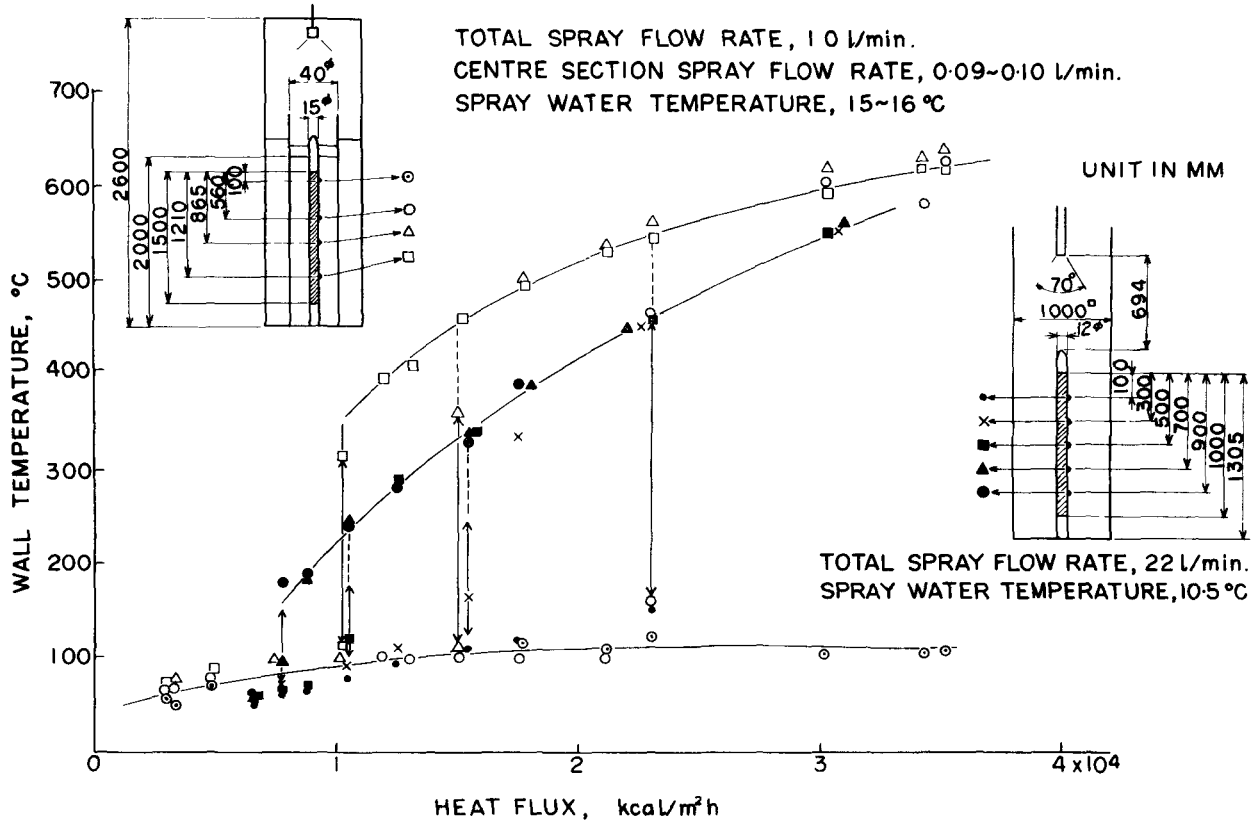
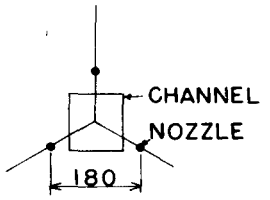
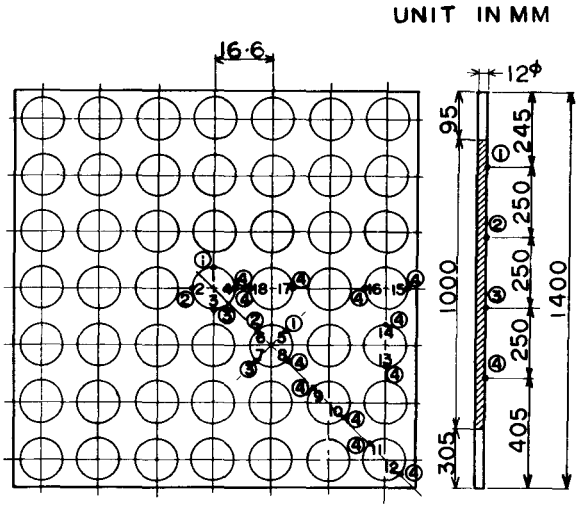


Figure 17. Cooling effect of spray (single heater)



1	2	3	4	5	6	7	← ROD POSITION
2·97	1·01	1·06	1·66	2·17	2·05	4·82	← AVERAGE PERCENTAGE FLOW
8	9	10	11	12	13	14	
1·44	2·65	1·57	1·44	1·86	1·82	1·64	
15	16	17	18	19	20	21	
2·40	1·47	1·19	1·29	1·47	1·87	1·77	
22	23	24	25	26	27	28	
1·96	1·67	1·07	1·30	1·28	1·50	1·63	
29	30	31	32	33	34	35	
2·33	2·02	1·51	1·30	1·09	1·61	1·11	
36	37	38	39	40	41	42	
3·13	2·00	1·98	1·57	1·64	2·00	2·73	
43	44	45	46	47	48	49	
5·42	4·01	4·40	2·51	1·28	1·40	5·30	

FLOW DISTRIBUTION TEST WITH NOZZLES LOCATED 640 mm ABOVE HANDLE.



THERMOCOUPLE ARRANGEMENT IN FUEL ASSEMBLY.

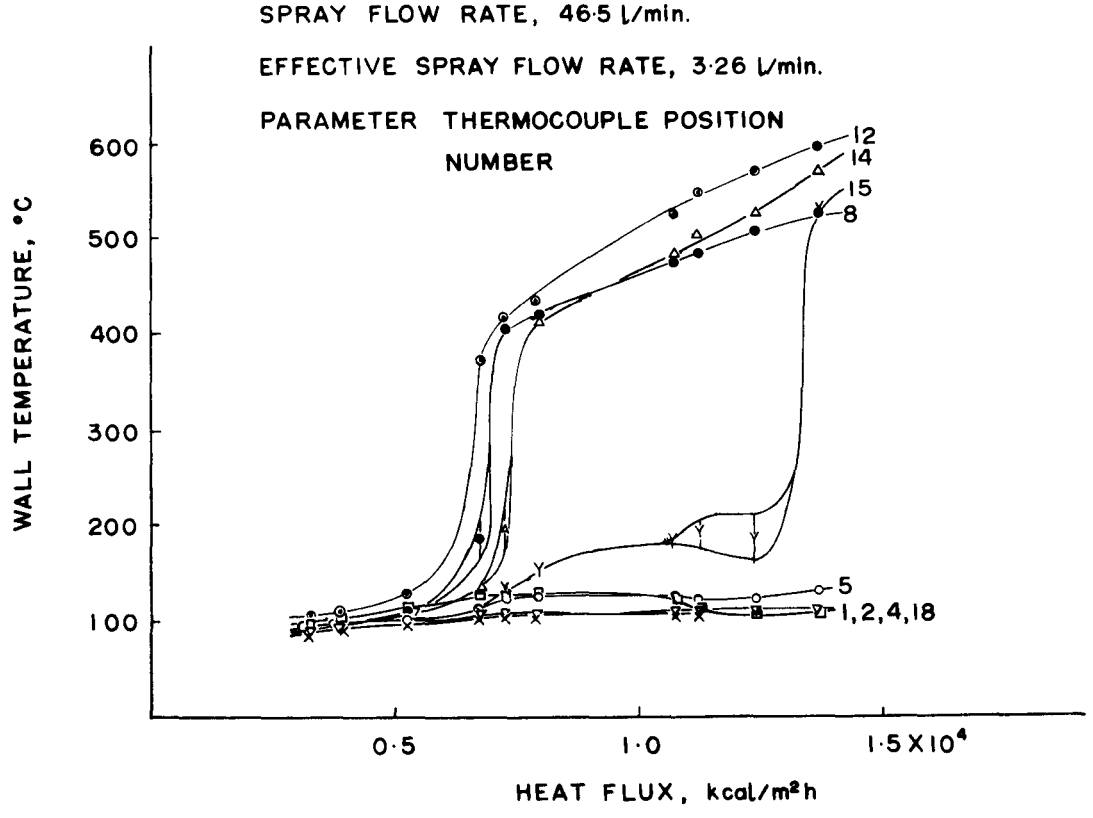


Figure 18. Water distribution and cooling effect of three-nozzle spray for multi-rod assembly

column assembly and 0.25 for the seven-row, seven-column assembly.

In the investigations into the cooling effect of the spray on the fuel element, both the single fuel rod and the fuel assembly have been used. The results obtained for several surface temperatures on the heater vs. mean heat flux are shown in Fig. 17 (single heater) and in Fig. 18 (assembly). In these experiments the water is sprayed downwards on the bar-type heater. In the case of the single heater, the high temperature parts which cannot be cooled effectively appeared at the lower part of the heater. The reason for this is that the water film, which flows down on the heater surface, gets completely evaporated before it reaches the lower part of the heater. In the case of the fuel assembly, because of the relative arrangement of the nozzles and the fuel rods in the assembly, all the fuel rods cannot be cooled uniformly, as shown in Fig. 18. The quantity of spray water for each part of the assembly, for the nonheating condition, is presented in Fig. 18. Water quantity on the wall side is large because in its measurement the quantity of water flowing down the wall was also included (water distribution for the heating condition has not been measured). The difference between the cooling effect of the spray in a bar-type heater (the heater is positioned concentrically in the insulated outer tube, and water is sprayed through the annular flow channel) and in a cylindrical heater (the pipe being insulated on the outer surface, heated on the inner surface and the spray flowing in the pipe), is represented in Fig. 19. The cooling effect is worse in the bar-type heater than in the cylindrical one because in the former case part of the spray flows down on the inner surface of the outer pipe. The heat output necessary for evaporating all the water is given by the point (*a*) in Fig. 19.

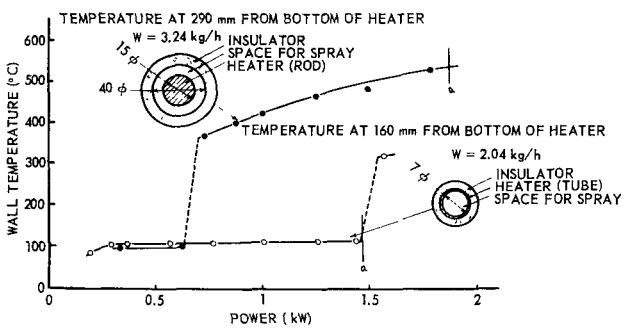


Figure 19. Cooling effect in bar-type and cylindrical heaters

Effectiveness of the container spray system

High pressure steam which arises in the container is cooled and condensed on the inner wall of the container or the auxiliary surfaces in the vessel. A series of basic experiments on condensation of the vapour containing noncondensable gas has been conducted at Hitachi Ltd. A mixture of steam and noncondensable gases (air, nitrogen, argon) is cooled on a vertical cooling surface (width × height = 140 × 300 mm), and the coefficients of condensing heat transfer obtained are shown in Fig. 20. It is found (*a*) that the condensing coefficient of heat transfer depends not on the

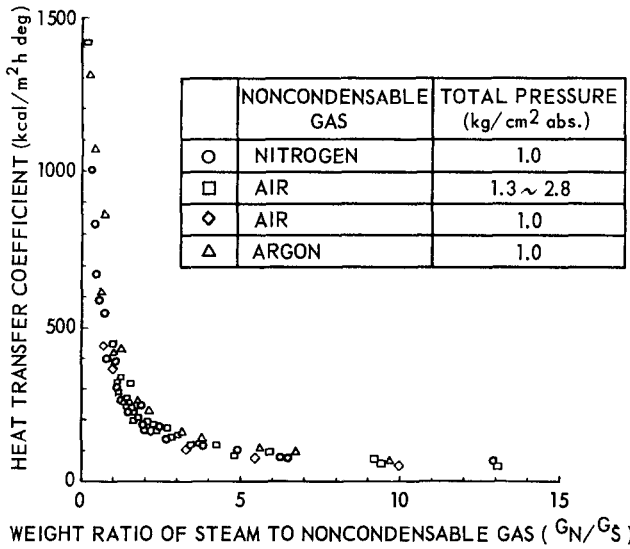


Figure 20. Result of condensation heat transfer of steam containing noncondensable gas

kind of gas but only on the weight ratio of the noncondensable gas, G_n, to the steam, G_s, and (*b*) that the coefficient decreases with the increase of the concentration of the noncondensable gas in the mixture.

The above-mentioned results are some of those obtained so far in the first phase of the research for project SAFE, and more detailed experiments are in progress.

Moreover, the methods of calculation for the theoretical prediction of the variation of pressure and temperature in the container for the loss-of-coolant accident; cooling effect and wash-out effect of the container spray on the radioactive gases; and the tightness at the penetrations of the pipes and cables into the container wall, are under study at Mitsubishi Atomic Power Industries Co. Ltd.

CONCLUSIONS

Major conclusions obtained through the theoretical evaluations are as follows:

(*a*) The F.P. gas release decreases with the increased spray flow rate; however, the JPDR design of 150 gpm will be adequate;

(*b*) A maximum of 5% of the F.P. gas release must be anticipated for JPDR, due to the nonuniform distribution of the core spray flow;

(*c*) The effect of nonuniformity of the fuel bundle spray flow distribution on the F.P. gas release is of secondary importance;

(*d*) The spray water drops must be greater than 0.4 mm in diameter to give stable spray cooling;

(*e*) Among various F.P. reduction factors the container shell effect is the largest, and the container spray wash-out effect is also important.

The experimental evaluations are now being conducted and final conclusions should be drawn when these are completed.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/436 Japon

Evaluation du système de refroidissement de réacteurs à eau légère après accident

par H. Uchida *et al.*

L'importance du système de refroisissement après accident est évaluée en considérant le phénomène dénommé « accident maximal vraisemblable » de réacteurs à eau légère.

Dans l'évaluation du système de vaporisateur du cœur, la non-uniformité de distribution du flux de vaporisateur du cœur a été prise en considération. Le calcul des pourcentages de ruptures de segment de combustible a été effectué en utilisant les données expérimentales sur la distribution du flux du vaporisateur du cœur ainsi que le transfert de chaleur du vaporisateur. Le calcul a été fait d'après les données du JPDR [BWR, 12,5 MW(e)].

On discute aussi la stabilité du refroidissement par vaporisateur. Dans l'évaluation du système de vaporisateur à réservoir, les facteurs de réduction des fuites de produits de fission ont été calculés et l'importance relative du système de vaporisateur à réservoir fait l'objet d'une discussion.

Pour évaluer, d'une manière expérimentale, l'efficacité du système de refroidissement après accident (vaporisateur du cœur, injection de sûreté et système de vaporisateur à réservoir), le projet SAFE (projet relatif aux installations et aux évaluations concernant la sûreté) a été élaboré par un comité de spécialistes pour la sécurité nucléaire du Japan Atomic Industrial Forum. Des spécialistes dans divers domaines (universités, laboratoires de recherche et fabricants) en sont membres. Dans la première étape d'exécution de ce projet (d'avril 1963 à septembre 1964), les quantités de réfrigérant écoulé dans les accidents de perte de réfrigérant ont été calculées par un modèle non nucléaire, ainsi que la température et la pression transitoires du réservoir. On envisage aussi d'exécuter des expériences de base sur les effets des divers types de buses de vaporisation et leur disposition, le transfert de chaleur du vaporisateur et l'effet de lavage pour comparer les résultats des calculs avec les données expérimentales. Les résultats expérimentaux obtenus jusqu'ici sont indiqués.

A/436 Япония

Оценка послеаварийных систем теплоносителя в энергетических реакторах с обычной водой

Х. Ушида *et al.*

По так называемой предельно возможной аварии в реакторах с обычной водой производилась оценка послеаварийной системы теплоносителя.

При оценке системы разбрызгивания жидкости в активной зоне была учтена неравномерность распределения потока жидкости. С использованием полученных экспериментально распределения потока разбрызгиваемой жидкости в активной зоне и распределения теплопередачи разбрызгиванием сделан количественный расчет процента разрывов оболочек тепловыделяющих элементов при авариях, связанных с утечкой теплоносителя. Расчет сделан по данным реактора JPDR [кипящий реактор на 12 *Мвт* (эл.)].

Обсуждаются также критерии стабильности системы охлаждения разбрызгиванием. При оценке системы разбрызгивания в баке реактора были вычислены коэффициенты уменьшения выброса продуктов деления, а также рассмотрено относительное значение систем разбрызгивания жидкости в баке.

Для экспериментальной оценки послеаварийных систем теплоносителя (систем разбрызгивания жидкости в активной зоне реактора, безопасного впрыскивания жидкости и систем разбрызгивания жидкости в баке реактора) комитетом специалистов по вопросам радиационной безопасности Японского атомного промышленного форума был утвержден проект SAFE (проект оценки безопасности и центра по оборудованию). В комитет входили различные специалисты (от университетов, научно-иссле-

довательских лабораторий и предприятий.) На первой стадии этого проекта (с апреля 1963 по сентябрь 1964 года) величина струи потока теплоносителя при аварии вследствие утечки теплоносителя была рассчитана на неядерной модели. То же относилось и к переходным изменениям давления в баке реактора. В задачу также входило проведение основных экспериментов для выяснения эффективности форсунок для разбрызгивания жидкости и мест их расположения, теплопередачи разбрызгиванием жидкости и эффекта вымывания с целью сравнения результатов расчетов и экспериментальных данных.

Приводятся полученные экспериментальные результаты.

A/436 Japón

Evaluación de los sistemas de refrigeración por aspersión para reactores de potencia de agua ligera

por H. Uchida et *al.*

Se estima la importancia de los sistemas de refrigeración de emergencia en relación con el llamado « accidente máximo previsible » de los reactores de potencia de agua ligera.

Se tiene en cuenta la distribución desigual del caudal de agua al estimar la eficacia del sistema de aspersión del núcleo. Se calculan los porcentajes de rupturas de cartuchos combustibles en el caso de accidentes por pérdida de refrigerante; los datos de partida son valores experimentales de la distribución del caudal de aspersión y de la transmisión del calor en estas condiciones. Los cálculos se hacen con los datos de proyecto del reactor JPDR (reactor de agua en ebullición de 12,5 MW(e)).

Se discuten también los criterios de estabilidad de la refrigeración por aspersión. Para estimar la eficacia del sistema de aspersión de la cubierta de contención, se calculan los factores de arrastre por lavado de los productos de fisión y se discute la importancia relativa de los sistemas de aspersión de la cubierta de contención.

Para evaluar experimentalmente los sistemas de refrigeración de emergencia (sistemas de aspersión del núcleo, inyecciones de seguridad y aspersión de la cubierta de confinamiento) se ha establecido el programa SAFE (safety assessment and facilities establishment project) dentro de una comisión de especialistas para el estudio de la seguridad nuclear del Japan Atomic Industrial Forum. Forman parte de esa comisión especialistas en materias diversas procedentes de las universidades, de los laboratorios de investigación y de las compañías suministradoras. Durante el primer período de este programa (desde abril de 1963 hasta septiembre de 1964), se han estimado los efectos de aspersión en varias cantidades para accidentes de pérdida de refrigerante, utilizando una maqueta sin combustible nuclear; también se han analizado las variaciones temporales de la temperatura y de la presión en el interior de una cubierta de contención. Para comparar los resultados teóricos con los experimentales, se intentan también llevar a cabo experimentos para determinar los efectos de distintas boquillas de aspersión con varias distribuciones, así como los efectos de la transmisión del calor y del lavado de los productos de fisión durante la aspersión. Se informa de los resultados obtenidos hasta la fecha.

An experimental study on the mechanical strength of fuel cladding tubes under loss-of-coolant accident conditions in water-cooled reactors

By Y. Mishima,[a] I. Katsura,[b] M. Kosaki,[c] K. Ono,[d] Z. Takao,[e] H. Tanaka,[f] M. Toyoda,[g] S. Wakisaka[h] and M. Yokosuka[i]*

In the event of a loss-of-coolant accident in a light-water reactor, a blow-out of primary coolant from a severed main pipe may cause a rapid fall in the system pressure; subsequently, even after the reactor has been scrammed, the fuel temperature will rise due to the decay heat and the reduction in heat-removal capacity. Provided that the internal pressure exerted by the fission gases on the fuel cladding is in equilibrium with the external system pressure, the net pressure difference acting on the cladding tube as a mechanical load will rise rapidly as the external pressure falls. Given these circumstances, two important problems need to be studied from the standpoint of reactor safety. One is whether the tube will remain intact under such a rapidly increasing load or not. The other is whether, even if the tube can withstand this change, the increase in the internal pressure caused by the temperature rise after the accident will provoke static rupture of the tube or not.

The purpose of the present study was to obtain useful information on these problems. The experiments were carried out in the following stages. As a first step, the loss-of-coolant accident was analysed to determine the experimental programme and fix the necessary parameters. The programme consisted of three different experiments. The first was a high-temperature tensile test (experiment A), in which the mechanical properties of cladding materials at elevated temperatures and their behaviour under abnormally high loading rates were examined. The second was a high-temperature burst test (experiment B), in which the static burst strength of the cladding tube was determined. The third was a dynamic pressure test (experiment C), designed to show the behaviour of cladding tubes under various conditions simulating actual accidents. However, it being very difficult to estimate the internal pressure exerted on a fuel-cladding tube by the fission gases since their rates of evolution has not yet been established with sufficient accuracy, this experiment was confined to determining the limiting pressure that would not provoke rupture even with the temperature and pressure changes encountered in accident conditions. The correlations between the results of these experiments were studied for both stainless steel and Zircaloy.

DETERMINATION OF PARAMETERS USED IN THE EXPERIMENTS

As typical current light-water reactors, the Yankee reactor was selected for the pressurized-water (PWR) type and the Dresden for the boiling-water (BWR) type. The calculation was carried out on a simplified model, and on the assumption that the coolant would be blown out at the speed of sound until the pressure dropped to a fairly low level. Figure 1 shows the calculated pressure plotted against time after an accident. In the case of a PWR, in the very first phase the sub-cooled water will blow out, the system pressure falling over a very short period at a rate of 560 kg/cm^2/s; thereafter, the blow-out material will change to a water-steam mixture associated with a comparatively slow rate of pressure drop. In the case of a BWR, the pressure will fall relatively slowly since blowing out will presumably start in the saturated state. The circumferential stress rate on the fuel-cladding tube during blow-out will rise to about 59 kg/mm^2 s in the case of PWR (initial stage), and to 0.17 kg/mm^2 s in the case of BWR. Although the tensile testing machines available at the time could measure the stress rate of the latter, they were not suitable for measuring that of

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the former. Accordingly, in experiment A the upper limit of the loading rate was fixed in the light of the capacity of the machines available. The fuel temperature will at first fall during blow-out of the coolant; then it will rise again as decay heat is generated, even if scramming is complete. The upper temperature limit was fixed somewhat arbitrarily at 700 °C for experiments A and B; this is higher than the actual cladding temperature in normal operation. However, experiment C was conducted over a cladding temperature range of 280 to 450 °C, which corresponds to actual operating conditions.

In experiment A, tensile specimens of AISI 304L and 347 type stainless steel and Zircaloy were tested. The dimensions of the cladding tube specimens used in experiments B and C are shown in Table 1: three values of D_o/t (where D_o is the outside diameter and t the wall thickness) were considered for each material; they are denoted by the symbols S, M and L respectively. The effects of hydrogen absorption up to 1 000 ppm on the mechanical strength were investigated on the Zircaloy-2 and Zircaloy-4 specimens.

Table 1. Nominal dimensions of tubular specimens

	347 stainless steel				Zircaloy-2			
	S	M	L	Yankee	S	M	L	Dresden
Outside diameter (od) (mm)	8.64	9.65	10.91	8.64	9.14	11.68	13.97	14.27
Wall thickness	0.51	0.44	0.38	0.53	0.76	0.76	0.64	0.76
(t) (mm) (od*t*) (mm) .	17.0	21.7	28.7	16.2	12.0	15.3	22.0	18.8

Table 2. Test parameters for tensile tests

Material	Loading rate (kg/mm²s)	Temperature (°C)
304 L stainless steel	0.3, 1.0, 3.0	RT,[a] 280, 360
	1.0	RT, 200, 280, 320, 360, 450, 550, 700
347 stainless steel	0.3, 1.0, 3.0, 5.0	RT, 280, 360
	1.0	RT, 200, 280, 320, 360, 450, 550, 700
Zircaloy-2	0.3, 1.0, 3.0	RT, 280, 360
	1.0	RT, 200, 280, 320, 360, 450, 550, 700

[a] RT = room temperature.

HIGH-TEMPERATURE TENSILE TEST (EXPERIMENT A)

The purpose of this experiment was to verify the effect of a relatively high loading rate on the mechanical properties, and to obtain data on the strength, of the materials at elevated temperatures. The combinations of loading rate and temperature used for each of the three materials tested are shown in Table 2.

The chemical composition and mechanical properties of the materials tested are listed in Table 3, and the shape and dimesions of typical tensile specimens are shown in Fig. 2.

From the test data, the mechanical properties, such as ultimate tensile strength, elongation and cross-sectional reduction over the temperature range room temperature to 700 °C, were calculated. The following conclusions were drawn:

(1) Effect of temperature

Figures 3 and 4 show the ultimate tensile strengths obtained from a statistical analysis of the effect of temperature and the test procedure followed, especially the loading rate. In these figures, T denotes 6 mm specimens and F 8 mm specimens. As expected, the effect of temperature on the mechanical properties was found to be very considerable, both for stainless steel and for Zircaloy. The ultimate tensile strength of stainless steel is characterized by a rather flat slope over the temperature range 300–500 °C (Fig. 3), whereas in the case of Zircaloy no such tendency is apparent (Fig. 4); and the fracture cross section changes significantly with temperature, being more or less square at lower temperatures and oblately ellipsoidal at higher temperatures. These are considered to be characteristic of metals of the hexagonal system.

(2) Effect of loading rate

The data relating to the mechanical properties of each material were statistically analysed, taking tempe-

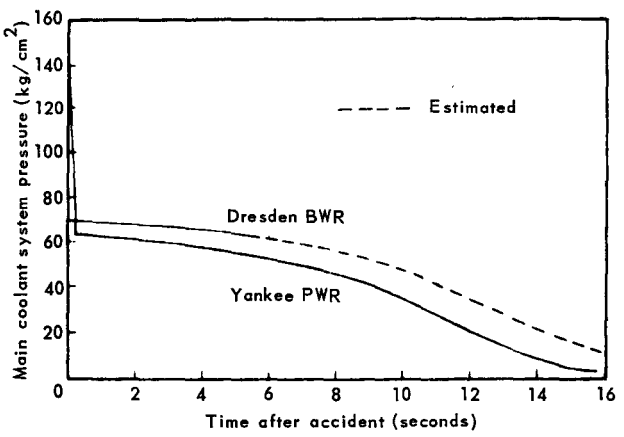


Figure 1. Change in main coolant system pressure following a loss of coolant accident

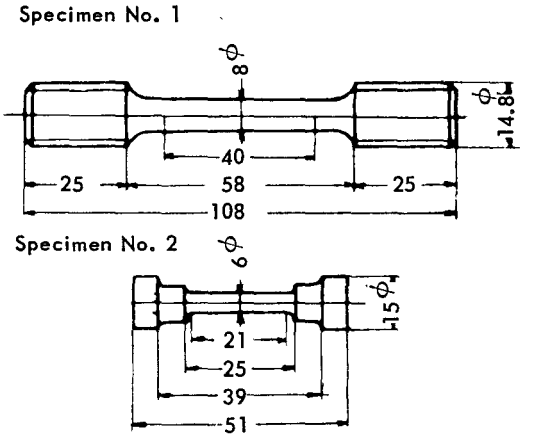


Figure 2. Tensile specimens

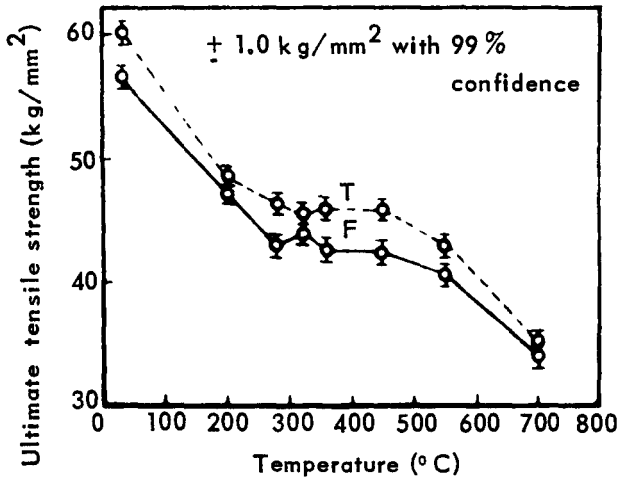


Figure 3. Ultimate tensile strength of 347 stainless steel at various temperatures

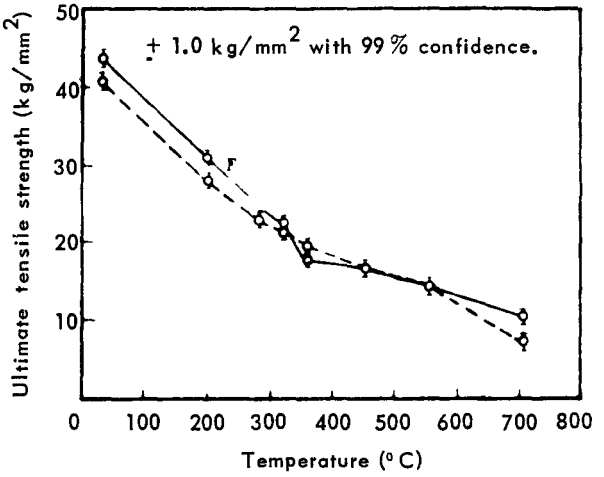


Figure 4. Ultimate tensile strength of Zircaloy-2 at various temperatures

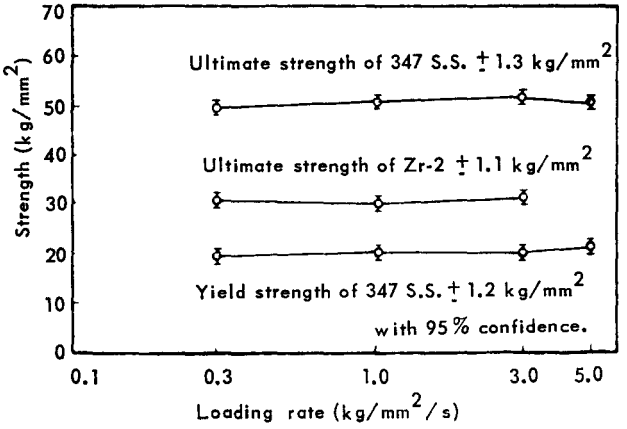


Figure 5. Strength at various loading rates

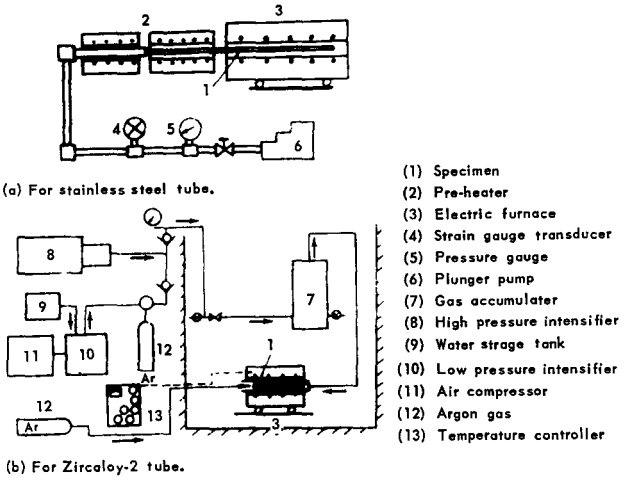


Figure 6. Arrangement of burst test apparatus

Table 3. Chemical composition and mechanical properties

(1) *Chemical composition* (wt%)

304 L SS

C: 0.021	Si: 0.46	Mn: 1.53
P: 0.021	S: 0.007	Ni: 12.00
Cr: 18.30	Nb + Ta: —	

347 SS

C: 0.06	Si: 0.66	Mn: 1.75
P: 0.025	S: 0.006	Ni: 12.30
Cr: 17.70	N + Ta: 0.80	

Zr-2

Sn: 1.39	Fe: 0.170	Cr: 0.099
Ni: 0.060	Al: 0.0056	C: 0.004
Si: 0.006	Hf: 0.007	H: 0.0015
N: 0.0032	O: 0.086	

(2) *Heat treatment*

304L SS	Annealed at 1 050 °C
347SS	Annealed at 1 100 °C
Zr-2	Annealed at 750 °C for 2 hr

(3) *Mechanical properties at room temperature*

	Ultimate tensile strength (kg/mm²)	Elongation (%)	Vicker's hardness (VHN 3 kg)
304L SS . .	64.5	52.0	136
347SS . . .	64.6	52.0	177
Zr-2	43.4	23.2	144

rature and loading rate into account, the effects of the latter being specially investigated. An example of its effects on the ultimate tensile strength is shown in Fig. 5, with the confidence limit specified. Similar investigations were carried out for elongation and cross-sectional reduction; from these it was concluded that the effect of loading rate on mechanical properties over the same temperature range is not clear for either material, except in the case of the yield point of 347 stainless steel, which tends to rise slightly with increasing loading rate.

HIGH-TEMPERATURE BURST TEST (EXPERIMENT B)

The purpose of this experiment was to pre-determine the static bursting strength of tube specimens before conducting the dynamic burst test (experiment C), and to obtain information about the effect of the rate of pressure increase on bursting strength at high temperatures, which could not be derived from experiment C. The test apparata are shown schematically in Fig. 6. Each has a hydraulic pump to pressurize the specimen, and a Bourdon-tube type, or strain-meter type, of pressure gauge from which the bursting pressure can be read off. In the case of Zircaloy, special precautions had to be taken to prevent surface oxidation of the specimen at high temperatures. A purging system was therefore added by which the air in the pressurizing system could be replaced by argon before testing was carried out. In this way, both the inside and outside surfaces of the specimen were kept in an atmosphere of argon while under test. The test parameters are given in Table 4.

Table 5 gives the chemical composition, and Table 6 the mechanical properties at room temperature, of the materials tested. The dimensions of the specimen tubes are to be found in Table 1, and their shapes and dimensions are depicted in Fig. 7.

As will be seen from the parameters selected, given in Table 4, efforts were concentrated on verifying the dependence of bursting strength on both temperature and dimensions (expressed by the ratio od/t). But a preliminary test revealed that the maximum rate of pressure increase that could be produced with the apparatus available had no effect on the bursting pressure. Therefore, the tests proper were limited to checking this fact, two tests only being conducted on stainless steel, applying pressure increase rates of 3 and 10 kg/cm² s respectively at 360 °C. The relationships between the bursting pressures and test temperatures obtained for the three different sizes of specimen in stainless steel and Zircaloy are shown in Figs. 8 and 9 respectively. In Fig. 8, the calculated bursting pressure

Table 4. Parameters for burst test

Material	Rate of pressure increase (kg/cm² s)	Temperature (°C)
347 SS	3.0	RT, 250, 320, 360, 450, 550 700
	10.0	360
Zr-2	0.5	RT, 280, 320, 360, 450, 550, 700
	5.0	320

Table 5. Chemical composition (average wt%)

347 SS

C: 0.06	Si: 0.57	Mn: 1.64
P: 0.025	S: 0.009	Ni: 12.06
Cr: 18.05	Nb: 0.74	

Zr-2

Sn: 1.39	Fe: 0.17	Cr: 0.099
Ni: 0.060	Al: 0.0056	C: 0.004
Si: 0.006	Hf: 0.007	H: 0.0035
N: 0.028	O: 0.071	

Table 6. Mechanical properties

	Ultimate tensile strength (kg/mm²)	2% off-set strength (kg/mm²)	Elongation (%)
347 SS	63.1	26.0	47.0
Zr-2	42.7	26.1	19.5

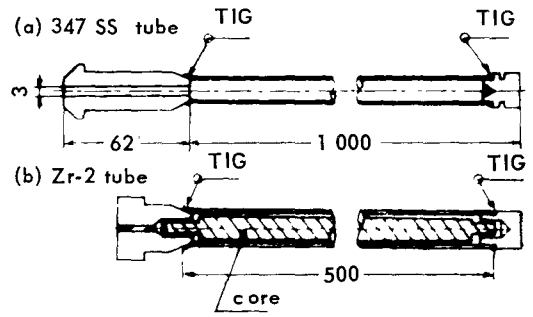


Figure 7. Burst test specimens

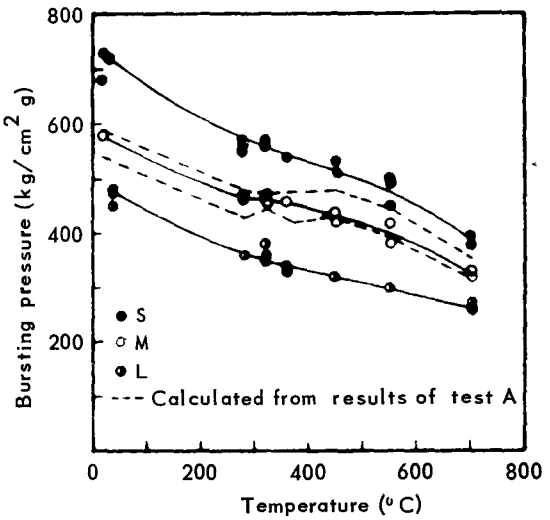


Figure 8. Bursting pressure of 347 stainless steel tube at various temperatures

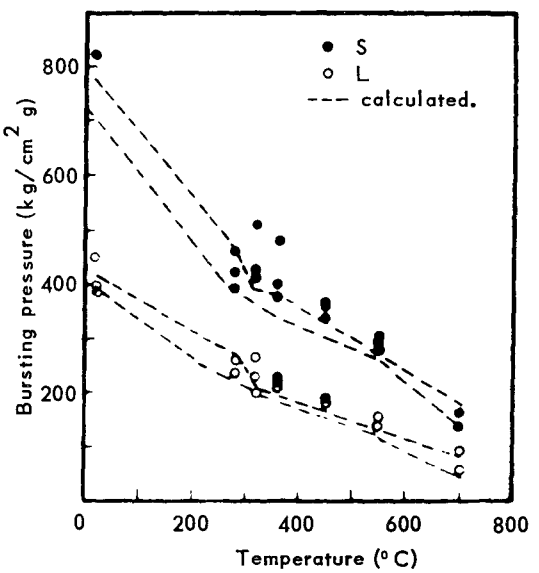


Figure 9. Bursting pressure of Zircaloy-2 tube at various temperatures

is also shown; this is derived from the formula $P = \sigma\, 2t/D_0$ using the ultimate tensile strength values obtained in experiment A (σ = ultimate tensile strength, D_0 = outside diameter, t = wall thickness). The measured and calculated bursting pressure values agree very well at temperatures from 280 to 360 °C, though some discrepancy appears close to room temperature and near 500 °C. Plate 1 shows the appearance of burst stainless-steel tubes. At room temperature, only an axial crack develops, resulting in a small hole, whereas near 300 °C the hole is much enlarged by the shearing forces acting on the ends of the crack. Above 450 °C, the specimen is fragmented where bursting occurs. Thus the aspect of the rupture depends on the test temperature.

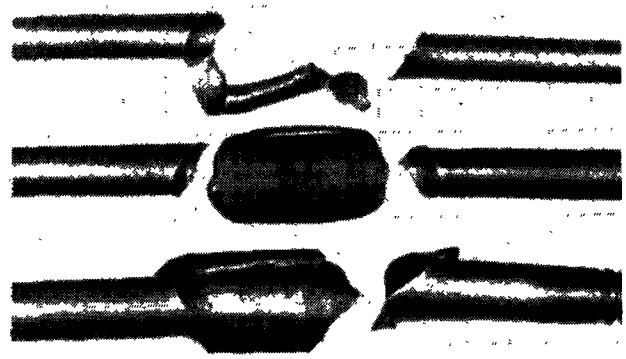
As in the case of stainless steel, different rates of pressure increase, namely, 0.5 and 5.0 kg/cm² s, were applied to Zircaloy at 320 °C only. No effect on the bursting pressure was observed. The aspect of the rupture depends to a considerable extent on the test temperature in the same way as with a stainless-steel tube.

DYNAMIC BURST TEST (EXPERIMENT C)

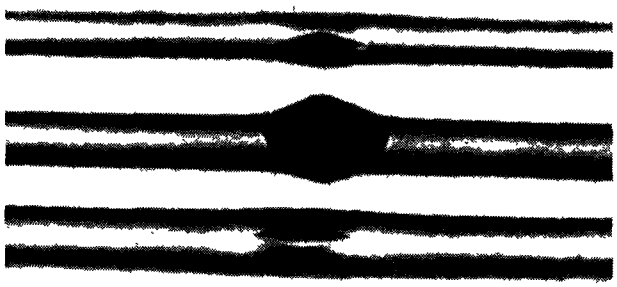
This experiment was the main feature of the study. It was designed to determine the bursting pressure of the cladding tube under the loss-of-coolant accident conditions predicted for PWR and BWR. Several problems were encountered in designing the test apparatus and in simulating actual accident conditions. The apparatus was designed in such a way that the external pressure on the tube could be abruptly decreased while maintaining high internal pressure and high surface temperature. Figure 10 shows the schematic diagram of the apparatus, and Plate 2 its external appearance. The main part is an autoclave simulating a reactor vessel, to which two high-pressure systems of 1 000 and 200 kg/cm² are connected independently; the appropriate instrumentation, e.g. for adjusting, controlling and recording pressure and temperature, is also provided. Each high-pressure system has an air-driven plunger-type pump (PP-1 and PP-2). The first can pressurize with water the inside of the specimen tube placed in the autoclave up to 1 000 kg/cm²; the second can fill the whole system with water and pressurize the autoclave itself up to 200 kg/cm². A rod-type sheathed electric heater is inserted into the specimen, thus simulating a fuel rod. The surface of the tube can be brought up to a pre-determined temperature, and the temperature inside the autoclave can be similarly controlled with the help of an electric furnace surrounding it. A primary pipe rupture can be simulated by quickly opening the diaphragm valve (AV-3) fitted in the autoclave outlet line, to which any size of orifice up to 8 mm can be fitted, thus making it possible to adjust the flow rate of the water blow-out and to maintain the rate of pressure decrease at a predetermined value. In such an experiment, accurate control and recording of the pressure and temperature are very important. The upper limits



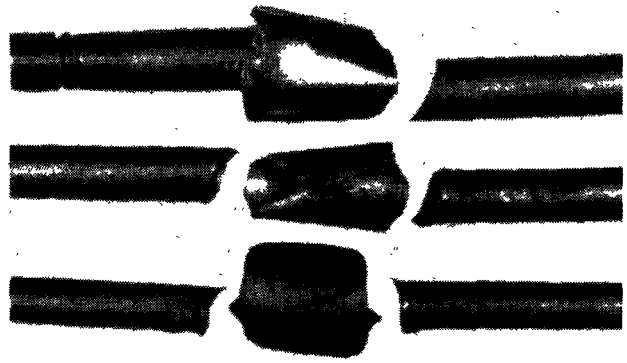
Temperature: 23 °C; Pressure increase rate: 3 kg/cm² s



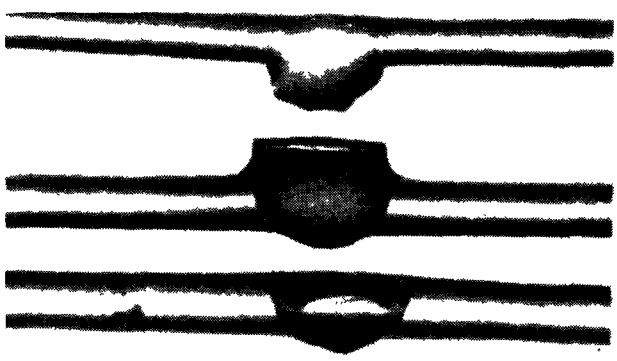
Temperature: 360 °C; Pressure increase rate: 10 kg/cm² s



Temperature: 280 °C; Pressure increase rate: 3 kg/cm² s



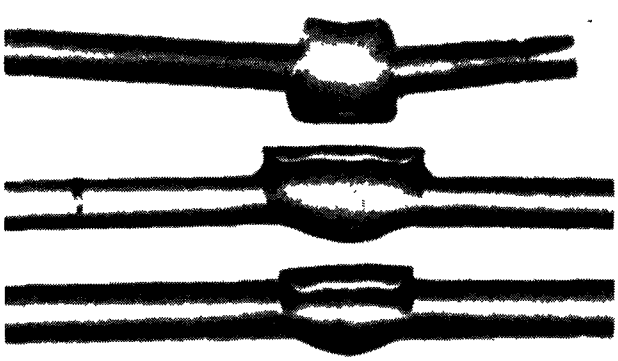
Temperature: 450 °C; Pressure increase rate: 3 kg/cm² s



Temperature: 320 °C; Pressure increase rate: 3 kg/cm² s



Temperature: 550 °C; Pressure increase rate: 3 kg/cm² s



Temperature: 360 °C; Pressure increase rate: 3 kg/cm² s



Temperature: 700 °C; Pressure increase rate: 3 kg/cm² s

Plate 1. Stainless steel specimens burst at various temperatures



Plate 2. External appearance of dynamic burst apparatus

of the two system pressures can be kept at any value within the over-all range by means of the pressure controllers (PC-1 and PC-2); the lower limits are fixed by adjusting the pressure of the air driving the two plunger pumps. The rapid pressure changes occurring in the two systems are measured by strain-meter-type pressure gauges (PH-1 and PH-2), and simultaneously recorded by a stylographic oscillo-graph (EP-3). The temperatures at the surface of the specimen and inside the autoclave are controlled and measured by sheathed alumel-chromel thermocouples connected to a thermostat-recorder. The water used is first demineralized, so that its electrical resistivity exceeds 1 MΩ cm.

The specimens tested were of 347 stainless steel, Zircaloy-2 and Zircaloy-4, with outside diameters and wall thicknesses as shown in Table 1, and 400 mm long. They were hermetically seal-welded at one end with a suitable plug. Figure 11 shows cross sections of the specimen. Figure 11(a) shows the stainless-steel specimen with its rod-type sheathed internal heater. As shown in the drawing, a conical copper ring is brazed to one end. When fitted to the autoclave head, this serves as a gasket for maintaining the pressure sealing and prevents the specimen from being ejected by axial thrust. The sheathed thermocouples are brazed to the surface of the stainless-steel tubes [Fig. 11(a)] or sweated on mechanically with a special copper ring in the case of Zircaloy-2 [Fig. 11(b)]. The chemical composition and mechanical properties at room temperature of the specimens were the same as in experiment B.

The test parameters applied are described in Table 7. Only one rate of pressure decrease was applied for each material, namely, PWR and BWR condition for stainless steel and BWR condition for Zircaloy. The surface temperature, however, was varied parametrically from 280 to 450 °C. To simulate the rate of external pressure decrease predicted for PWR, it was unnecessary to provide an orifice on the autoclave outlet line. It took from 0.13 to 0.20 s to reach saturation from the beginning of blow-out. The variations in this time were assumed to depend mainly on the boiling conditions prevailing on the surface of the specimen; for example, if film boiling occurred, the rate of pressure decrease was relatively slow. Under BWR conditions, the pressure fell relatively slowly because blow-out of a water-steam mixture starts from the initial state at 70 kg/cm^2 and 285 °C. Therefore, with a 5 mm orifice, an initial rate of decrease of 2.0 kg/cm^2 s was attained. A preliminary test revealed that when blow-out began the surface temperature fell, due to the sudden inception of surface boiling. An additional protective sleeve around the specimen by way of thermal insulation was needed.

The test procedure was, briefly, as follows:

After a specimen and its protective sleeve had been fitted to the autoclave head, the autoclave was tightly closed. As the first step, the interal and external pressures and temperature of the specimen were simultaneously brought up to the pre-determined values, with the inter-connexion valves of both systems open, by means of the plunger pump (PP-2) and the autoclave heating furnace. The valves were then closed and the two systems isolated. The internal pressure on the specimen was then increased up to the value of the static bursting pressure, plus several tenths of the external pressure, according to the dimensions of the specimen, by means of pump PP-1. (The static bursting pressure of a similar specimen had already been determined in experiment B.) At the same time, the surface temperature was increased to a pre-determined value by means of the sheathed heater fitted inside the specimen. When equilibrium was reached, the diaphragm

Table 7. Parameters for dynamic burst test

	Initial external pressure (kg/mm^2)	Initial external temperature (°C)	Rate of external pressure fall (kg/mm^2 s)	Surface temperature of specimen (°C)
347 SS	140	280	Within 0.25 s to saturation	280 (285) 320 360 450
Zr-2	70	285	2.0	

Table 8. Effect of absorbed hydrogen; parameters for tensile test

Factory	Material	Loading rate (kg/mm² s)	Hydrogen content (ppm)	Temperature (°C × 15 minutes) keep	Property
S	Zr-2 Zr-4	0.3 3.0	As received 100 200 300 400 500	RT 300	Ultimate tensile strength yield strength (0.2%) elongation
F	Zr-2 Zr-4	0.3 3.0	As received 300	RT 200 300 350 400	Tensile strength elongation

valve AV-3 was opened. This was followed by an abrupt drop in the external pressure and rupture of the specimen. The temperatures and pressures were continuously recorded during this phenomenon. Sometimes the specimen burst at a relatively low pressure during pressurizing, but this could be attributed to a defect in the specimen, and such tests were accordingly discarded. Plate 3 shows a typical pressure-time record at the moment of rupture when testing an M-type stainless-steel specimen. At the beginning of this test, the external and internal pressures were 140 kg/cm²(g) and 580 kg/cm²(g) respectively, and the external and surface temperatures 280 °C. The specimen burst in a very short time during the sub-cooled water blow-out so that saturation was not reached. It is assumed that this burst was caused by a high rate of increase in the internal pressure provoked by the abrupt fall in the external pressure. In some tests, bursting occurred while the external pressure was falling relatively slowly, maintaining saturation conditions.

The results of the experiments can be summarized as follows:

In Fig. 12, the net pressure when a 347 stainless-steel specimen burst is plotted against the temperature, the average static bursting pressures obtained in experiment B being added for comparison. Both values show a considerable measure of agreement for each size of specimen. As already mentioned, the rate of decrease of external pressure at the moment of bursting varied from about 400 kg/cm² s (maximum during sub-cooled blow-out) to a relatively slow figure after saturation had been reached. This information made it possible to study the dependence of tube strength on the rate of decrease of the external pressure. The relationship between the average value of this rate and the nominal stress derived from $P.\ D_0/2t$ was discussed. This expression makes it possible to exclude the effect of dimensional differences. From this experiment, it was concluded that at 280 °C the bursting stress tends to rise a little more sharply than a rate of between 300 and 400 kg/cm² s. But, generally speaking, the bursting stress is not affected so much at such a high loading rate as was achieved in these experiments; moreover, it does not differ greatly from the static value obtained in experiment B.

As shown in Plate 4, the starting conditions with a Zircaloy-2 specimen were: external pressure, 70 kg/cm²(g); temperature, 285 °C (saturation); surface temperature, 440 °C; and internal pressure, 680 kg/cm²(g). The rate of decrease of the external pressure was 2.3 kg/cm² s. The net bursting pressures recorded are plotted against the corresponding temperatures in Fig. 13. The static and dynamic values agree very well. It was thus confirmed that, as in the case of stainless steel, the rate of decrease of external pressure has no effect on the mechanical properties. Moreover, the bursting stress calculated from the expression $P.D_0/2t$ falls within the range of the ultimate tensile strength values obtained in experiment A, with satisfactory agreement.

EFFECT OF ABSORBED HYDROGEN

Tensile tests and static and dynamic burst tests were carried out on Zircaloy-2 and Zircaloy-4 to investigate the effect of absorbed hydrogen on their mechanical properties.

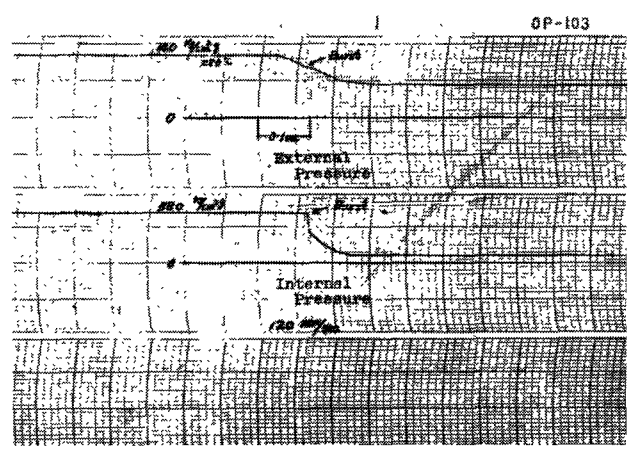


Plate 3. Typical pressure time recording for stainless steel tube

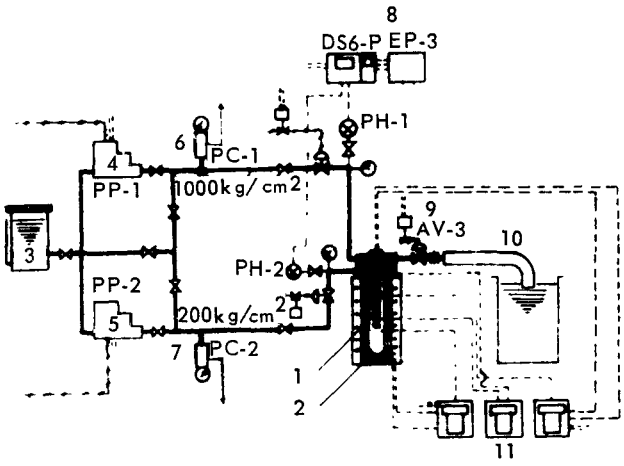


Figure 10. Schematic flow diagram of dynamic burst test

1: Specimen; 2: Autoclave; 3: Water storage tank; 4: High pressure plunger pump; 5: Low pressure plunger pump; 6: High pressure release valve; 7: Low pressure release valve; 8: Pressure recorder; 9: Discharge valve; 10: Discharge line; 11: Temperature recorders

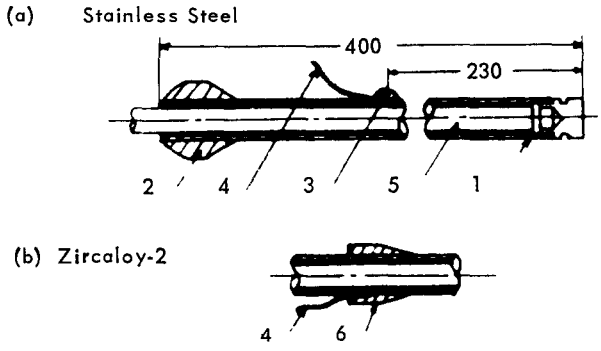


Figure 11. Specimens for dynamic burst test

1: Specimen; 2: Copper packing; 3: Brazing; 4: Thermocouple; 5: Sheath heater; 6: Copper ring

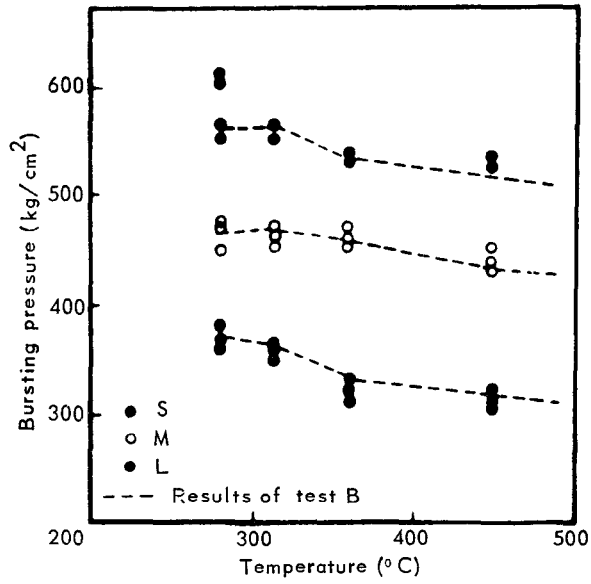


Figure 12. Bursting pressure of 347 stainless steel tube at various temperatures

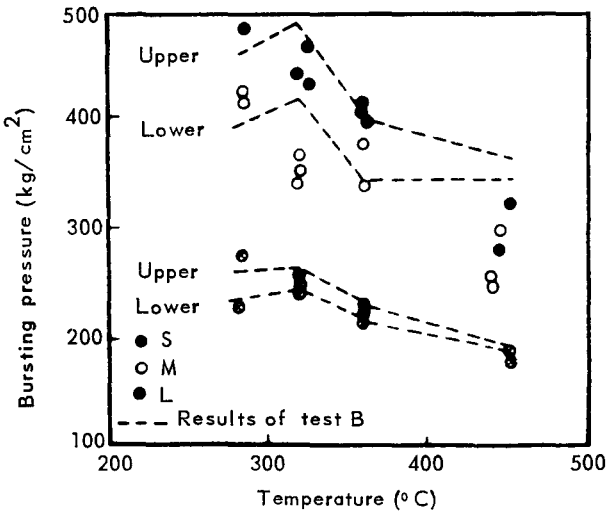


Figure 13. Bursting pressure of Zircaloy-2 tube at various temperatures

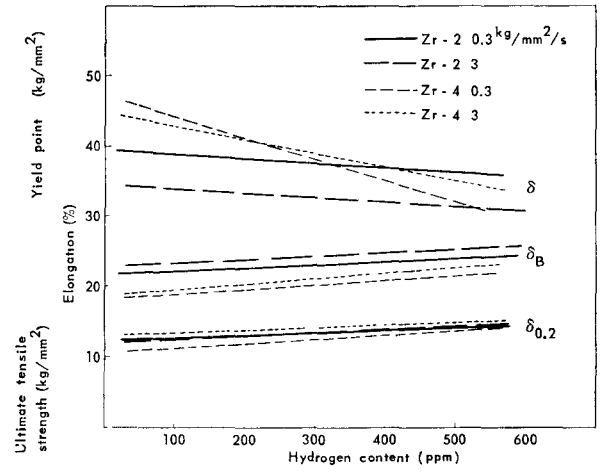


Figure 14. Effect of hydrogen content on tensile properties of Zircaloy-2 and -4 at 300 °C

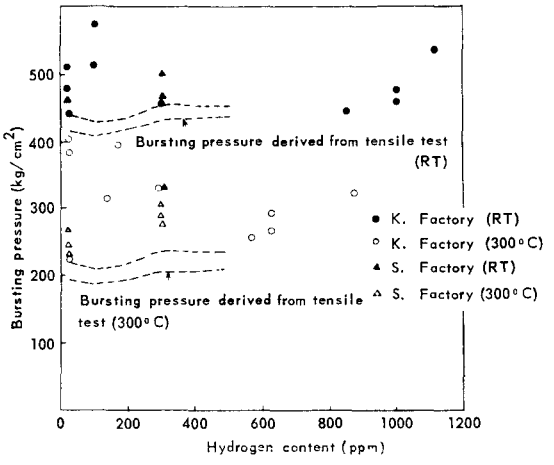


Figure 15. Effect of hydrogen content on bursting pressure of Zircaloy-2 tubes made in Japan

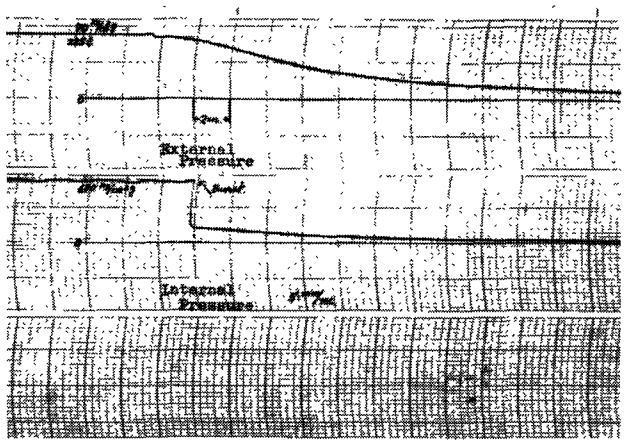


Plate 4. Typical pressure time recording for Zircaloy-2 tube

Tensile test. The specimen was of the plate type: 1 mm thick, 20 mm wide and 200 mm long. Two separate testing machines were used. The test parameters were as shown in Table 8. The ultimate tensile strength (σ_B), elongation (δ) and, for some specimens, yield point ($\sigma_{0\cdot2}$) were determined and summarized. The results showing the effect of absorbed hydrogen on $\sigma B'$ $\sigma 0.2'$ and δ at 300 °C for two loading rates are given in Fig. 14. Consideration of these results suggests that it is safe to conclude that up to 300 ppm absorbed hydrogen has no appreciable effect on the tensile strength of Zircaloy-2 or Zircaloy-4 regardless of increasing temperature and loading rate.

Static burst test. Static burst tests were carried out on Zircaloy-2 tubes fabricated respectively in Japan and USA to investigate the effects of absorbed hydrogen and temperature on the bursting strength, the content of hydrogen being run up to 1 000 ppm and the temperature up to 400 °C.

The dimensions of the specimens were: 13.97 mm od and 0.635 mm wall thickness for the tube made in Japan; and 12.70 mm od and 0.686 mm wall thickness for the tube imported from USA. The tubes were seal-welded at one end with a suitable plug, and had two different effective lengths: 500 and 300 mm. A suitable core was inserted into the tube to reduce the volume of water. The test parameters are given in Table 9.

The relationships between absorbed hydrogen and bursting pressure revealed by the test are shown in Fig. 15 and Fig. 16 (for Japanese and USA tubes respectively). Only one value is given in Fig. 16 for the calculated pressure derived from the ultimate tensile strength of the plate specimen. The figures show that: (*a*) the static bursting pressure is unaffected either at room temperature or at higher temperatures, even when the amount of absorbed hydrogen is as high as 1 000 ppm; (*b*) at higher temperatures, the bursting pressure decreases, the rate of decrease corresponding well with the fall in the value of the ultimate tensile strength; (*c*) in general, the actual bursting pressure is slightly higher than that calculated from the equation $P = \sigma B.2t/D_0$ based on the ultimate tensile strength determined in the tensile test; and (*d*) these tendencies are common to both Japaneses and USA tubes. However, the latter have a considerably higher bursting

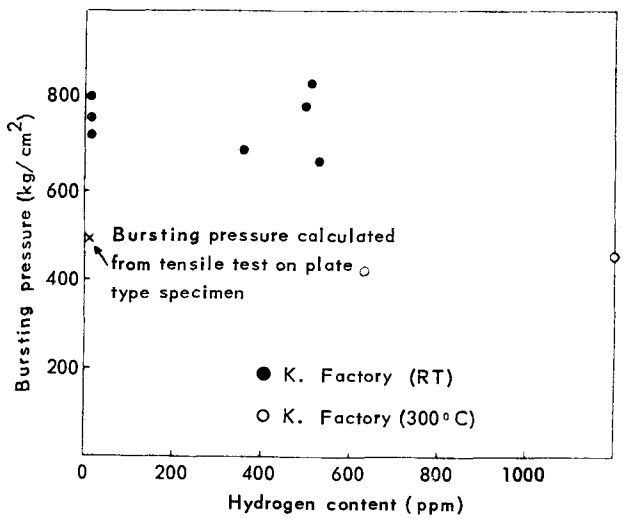


Figure 16. Effect of hydrogen content on bursting pressure of Zircaloy-2 made in the USA

pressure than the former; this is a consequence of the slightly higher ultimate tensile strength of the USA tubes, due to their smaller diameter, greater wall thickness and higher oxygen content.

Table 9. Effect of absorbed hydrogen; parameters for static burst test

Factory	Material	Hydrogen content (ppm)	Temperature (0 °C)	Rate of pressure increasing (kg cm² s)
S	Zr-2 tube made in Japan	As received	RT	3
		300	200	
			300	
			350	
			400	
F	Zr-2 tube made in Japan	As received	RT	0.5
	Made in USA	≈ 1 000	300	

Dynamic burst test. To investigate the effect of absorbed hydrogen on the bursting pressure of Zircaloy-2 tubes, the same tests as in experiment C were performed. The tubes, which were pressurized internally, were subjected to simulated loss-of-coolant conditions predicted for BWR. The bursting pressure, the external pressure and the temperature, as functions of elapsed time, were determined with a view to estimating the actual bursting strength under such accident conditions. The shape and length of the specimens, the test procedure and the measuring methods were the same as in experiment C. The starting conditions were: external pressure, 70 kg/cm^2; and external temperature, 285 °C. The rate of pressure decrease was kept constant during the test at 2.0 kg/cm^2 s. The hydrogen absorbed was limited to 300 ppm and the surface temperatures of the specimens were set at 300, 350 and 400 °C. The same Japanese and USA tubes were used as in the static burst test.

A typical test record for the USA tube is shown in Plate 5. The internal pressure was 500 kg/cm^2 and the surface temperature 300 °C. The test data obtained in

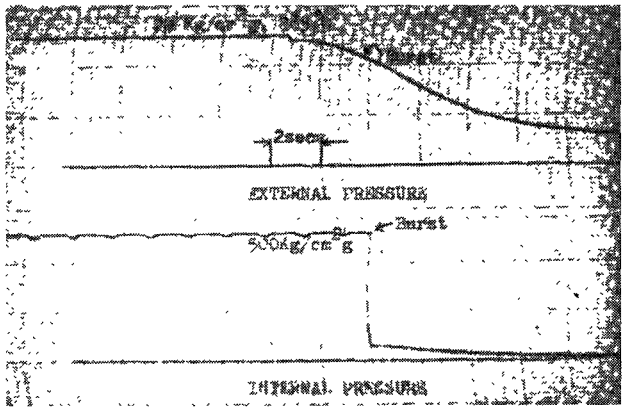


Plate 5. Typical pressure time recording for Zircaloy-2 tube (USA) under dynamic burst test

each test are summarized in Table 10. Although the data obtained from the dynamic burst test spread over a rather wide range, the bursting pressures obtained by the static test fall within virtually the same range. The figure obtained experimentally is a little higher than that calculated from the ultimate tensile strength. Generally speaking, hydrogen absorbed up to the amount used in the test has not much influence on the bursting pressure.

CONCLUSIONS

The experimental studies described lead to the following general conclusions.

(*a*) It is expected that in the case of a loss-of-coolant accident in PWR, which might be caused by a rapid rise in the net internal pressure due to an abrupt fall in the external pressure in the initial stage of blow-out, the bursting pressure would be slightly higher than the static bursting pressure. This means that the rapid increase in loading rate associated with such an accident does not greatly affect the bursting pressure. However, the experiments do provide some evidence to support the view that higher ultimate tensile strengths and yield points are obtained with the quickest loading rates, as in the impact tensile test by explosion.

(*b*) The data obtained in experiment A on high-temperature strength under rapidly increasing loading rates are very useful, even though the rate itself was not extremely high, because hitherto this kind of information has rarely been obtained experimentally. Some degree of directionality in tensile properties has been reported generally in rolled-plate type specimens, and a similar difference is to be expected between linear and tangential tensile strength in tube-type specimens. It was found, however, that the tube bursting pressure could be predicted theoretically with sufficient accuracy from the expression $P = \sigma 2t/D_o$, where σ is the ultimate axial tensile strength determined by an ordinary tensile test. Moreover, there is very little difference between the bursting pressures obtained by dynamic and static tests respectively.

(*c*) It was observed in experiment B that the bursting pressure fell to an extremely low level when there were flaws in the external or internal surface of the cladding tube. It follows that it is very important to carry out a complete, non-destructive inspection of the tube to ensure that it is in perfect condition for the burst tests. The dimensional tolerance of cladding tubes is also important, since dimensional variations affect the bursting pressure.

(*d*) Although the phenomena which would follow a loss-of-coolant accident were studied in these experiments, the investigations were limited to the effect of pressure and temperature changes; the metallurgical and chemical effects, however, and especially those of corrosion and chemical reaction on Zircaloy, also merit attention from the viewpoint of accident analysis or material evaluation. An in-pile irradiation experiment is therefore planned to check the metallurgical behaviour of cladding tubes under neutron irradiation.

(*e*) Up to about 1 000 ppm, the absorbed hydrogen content of Zircaloy, as tested in the present study, has no definite effect on the ultimate tensile strength of the material or on its bursting pressure, though it may be assumed that it will have some effect on impact properties.

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Table 10. Effect of absorbed hydrogen: results of dynamic burst test

Hydrogen content (ppm)	Number	Temperature (°C)	Outside diameter (mm)	Wall thickness (mm)	Internal pressure (at bursting) (kg/cm²)	External pressure (at bursting) (kg/cm²)	Bursting pressure Po (kg/cm²)	Time to bursting (s)	Rate of pressure increase (kg/cm² s)	Bursting pressure derived from tensile test (kg/cm²)
	[a] K-9	300	13.93	0.62	300	100	200	—	—	
	7	300	13.95	0.64	280	36	244	6.2	5.5	
As received (21)	8	300	13.94	0.63	260	25	235	9.2	4.9	221 ~ 193
	15	300	13.89	0.63	260	5	255	—	—	
	18	300	13.91	0.62	260	41	219	5.6	5.2	
	K-6	350	13.93	0.64	240	15	225	14.0	3.9	193 ~ 186
	K-1	400	14.00	0.65	280	20	260	11.0	4.5	
	2	400	13.99	0.63	260	40	220	16.0	3.1	
	K-12	300	13.93	0.63	260	10	250	44.6	1.3	
300	13	300	13.95	0.65	280	13	267	22.8	2.5	236 ~ 202
	14	300	13.92	0.63	300	21	279	9.4	5.2	
	16	300	13.95	0.64	280	20	260	11.0	4.5	
	K-1	300	12.68	0.73	520	70	450	—	—	
As received (11)	2	300	12.68	0.73	500	60	440	3.2	3.1	
	3	300	12.67	0.72	500	65	435	1.6	3.1	
	[b] R-5	300	12.67	0.74	470	32	438	6.8	5.6	
	6	300	12.68	0.72	460	39	421	12.0	2.6	
300	7	300	12.68	0.73	500	70	430	—	—	
	1	400	12.64	0.69	390	13	377	21.8	2.6	
	2	400	12.64	0.71	370	18	352	17.1	3.1	

[a] K: Zircaloy-2 made in Japan; [b] R: Zircaloy-2 made in USA.

ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/438 Japon

Etude expérimentale de la résistance mécanique des gaines de combustible en cas de perte accidentelle de réfrigérant dans les réacteurs à eau légère

par Y. Mishima *et al.*

Le but de cette expérience exécutée par un groupe de spécialistes sous les auspices du Japan Atomic Industrial Forum était d'examiner et de confirmer l'intégrité mécanique et le comportement des gaines de combustible de réacteurs à eau légère en cas de perte accidentelle de réfrigérant.

L'accumulation, dans certaines limites, de gaz de fission à l'intérieur des éléments de combustible ne constitue pas une source de pression interne influençant les gaines de combustible pendant le fonctionnement du réacteur, puisque cette pression est compensée par celle du réfrigérant à l'extérieur des gaines. Si toutefois la pression du réfrigérant tombe pour une raison telle qu'une perte accidentelle de réfrigérant, il se produit instantanément une différence de pression effective entre l'intérieur et l'extérieur des gaines, ce qui peut entraîner une rupture. La hausse de température provoquée par la chaleur de désintégration après la perte de réfrigérant augmente la pression interne, ce qui peut également donner lieu à une rupture. L'expérience en question a permis d'évaluer le coefficient de sécurité des gaines dans des conditions simulées de perte de réfrigérant.

On a choisi pour les essais des tubes en acier inoxydable AISI 304L et 347 et des tubes en Zircaloy-2, avec un rapport diamètre/épaisseur compris entre 12,0 et 28,7. Des échantillons de tubes en Zircaloy-2 contenant de l'hydrogène absorbé ont également été choisis en vue d'évaluer l'effet de l'absorption d'hydrogène sur le comportement de la gaine du point de vue mécanique en cas de perte accidentelle de réfrigérant.

Les conditions expérimentales ont été déterminées d'après l'analyse des cas de perte accidentelle de réfrigérant pour des réacteurs typiques sous pression ou à eau bouillante. Les expériences ont été exécutées dans les conditions suivantes:

a) Essai de traction des matériaux de gainage aux températures élevées allant jusqu'à 700 °C, et aux divers taux de charge normaux allant jusqu'à 5,0 kg/mm² s;

b) Essais d'éclatement des gaines de différentes dimensions à des températures allant de la température ambiante à 700 °C;

c) Pour l'essai d'éclatement dynamique dans les différentes conditions de perte accidentelle de réfrigérant, les échantillons étaient soumis à la pression interne fixée à l'avance, à des températures de surface allant de 280 à 450 °C et sous une pression externe correspondant à celle du réfrigérant, après quoi la pression externe était brusquement réduite au taux prévu en cas de perte accidentelle de réfrigérant.

Une série d'expériences effectuées dans ces conditions ont permis de conclure que les caractéristiques mécaniques des gaines sont à peine influencées par le taux de charge correspondant aux taux de chute de pression prévu en cas de perte accidentelle de réfrigérant, et ont confirmé que la pression de rupture de la gaine de combustible peut être estimée d'après la résistance des matériaux à la traction. On a observé une expansion moins forte, après rupture, de la gaine en Zircaloy-2 qui avait absorbé 300 ppm d'hydrogène environ, alors que sa résistance à la traction n'avait presque pas changé.

A/438 Япония

Экспериментальное исследование механической прочности трубчатых оболочек твэлов в условиях аварийной утечки теплоносителя из первичного контура водяного реактора

И. Мишима *et al.*

Цель этого эксперимента, который был проведен группой специалистов под руководством Японского атомного промышленного форума, состояла в изучении и подтверждении механической целостности и свойств трубчатых оболочек твэлов для реакторов с обычной водой в условиях аварийной утечки теплоносителя.

Накопление до известной степени газообразных продуктов деления внутри твэлов не оказывает влияния на их трубчатые оболочки во время работы реактора, так как это накопление не может являться источником внутреннего давления, поскольку давление теплоносителя снаружи труб твэлов компенсирует это внутреннее давление. Если давление теплоносителя падает в результате, например, его аварийной утечки, то существенная разность между внутренним и внешним давлениями на трубчатые оболочки мгновенно возрастает, что может привести к разрыву. Возрастание температуры после утечки теплоносителя вследствие остаточных тепловыделений вызывает повышение внутреннего давления, которое также приводит к разрыву. В ходе эксперимента была проведе-

на оценка надежности оболочки твэла при имитации таких условий.

Для испытаний были выбраны трубы из нержавеющих сталей типа AISI 304L, 347 и из циркалоя-2 с соотношениями их диаметра к толщине в пределах от 12,0 до 28,7. Были также выбраны наводороженные образцы труб из циркалоя-2 для оценки влияния поглощения водорода на механические свойства при аварийных условиях.

Экспериментальные условия определялись анализом аварийных утечек теплоносителя в типичных водяных реакторах как под давлением, так и кипящего типа.

Эксперименты проводились в следующих условиях:

a) испытания на растяжение материалов оболочек при высоких температурах до 700° С и при различных умеренных значениях скоростей приложения нагрузки до 5,0 *кг/мм²·сек;*

b) испытания на разрушение трубчатых оболочек различных размеров в диапазоне температур от комнатной до 700° С;

c) испытания на динамическое разрушение в различных условиях аварийной утечки теплоносителя, когда образцы подвергались внутреннему давлению до заданных значений при температурах поверхности от 280 до 450° С и внешнем давлении, соответствующем давлению теплоносителя, после чего внешнее давление внезапно снижалось до уровня, предполагаемого для аварийной утечки теплоносителя.

На основе вышеуказанной серии экспериментов был сделан вывод о том, что на механические свойства трубчатых оболочек практически не влияет предполагаемая скорость уменьшения давления при аварийной утечке теплоносителя. Экспериментальным путем была подтверждена возможность оценки разрушающего давления для трубчатых оболочек твэлов на основе данных по сопротивлению разрыву материалов. В трубах из циркалоя-2, наводороженных до 0,03%, наблюдалось меньшее расширение после разрыва, хотя и было установлено, что их сопротивление разрыву изменилось мало.

A/438 Japón

Estudio experimental sobre la resistencia mecánica de vainas de combustible en el caso de un accidente de pérdida de refrigerante en reactores refrigerados por agua

por Y. Mishima *et al.*

El propósito de este estudio, que ha sido llevado a cabo por un grupo de especialistas bajo los auspicios del Japan Atomic Industrial Forum, es examinar y confirmar la integridad mecánica y comportamiento de los tubos de envainado de combustibles para reactores de agua ligera, en el caso de un accidente de pérdida de refrigerante.

La presión interna a que da lugar una acumulación limitada de gases de fisión dentro de los elementos combustibles no afecterá a las vainas mientras el reactor esté funcionando, puesto que el refrigerante exterior compensa dicha presión. Sin embargo, si la presión del refrigerante bajase por alguna razón, tal como la del accidente en estudio, aparecería instantáneamente una diferencia efectiva entre las presiones interna y externa, de lo cual podría resultar la ruptura de la vaina. La elevación de temperatura debida al calor residual después de la pérdida de refrigerante da lugar a su vez a una elevación de la presión interna, que puede también ocasionar la ruptura. Este experimento se efectuó simulando dichas condiciones, lo que permitió comprobar la seguridad de la vaina del combustible.

Se seleccionaron para las pruebas tubos de acero inoxidable AISI 304L y 347, y Zircaloy-2, con una relación de diámetro a espesor entre 12 y 28,7. También se utilizaron muestras de Zircaloy-2 con hidrógeno absorbido, para estudiar la influencia de la citada absorción en el comportamiento mecánico de los tubos sometidos a las condiciones del accidente.

Las condiciones experimentales se establecieron a partir del análisis de accidente de pérdida de refrigerante para reactores típicos de agua a presión y en ebullición. Estas condiciones, bajo las cuales se realizaron los experimentos, fueron las siguientes:

a) Ensayo estático de resistencia de los materiales de envainado a altas temperaturas, hasta 700 °C, y con unas velocidades moderadas de aplicación de la carga de hasta 5,0 kg/mm² s.

b) Prueba de ruptura de tubos de envainado de varias dimensiones y para temperaturas comprendidas entre la temperatura ambiente y los 700 °C.

c) En el ensayo dinámico de ruptura, bajo las diversas condiciones del accidente de pérdida de refrigerante, las muestras se sometieron a una presión interna predeterminada y temperaturas superficiales comprendidas entre 280 y 450 °C, siendo la presión exterior la correspondiente al refrigerante. Entonces, dicha presión exterior fue repentinamente reducida según la velocidad de variación prevista para el caso del accidente.

La conclusión de esta serie de experimentos es que las propiedades mecánicas de los tubos de envainado resultan muy poco afectadas por la variación de carga que resulta de la velocidad de decrecimiento de la presión prevista para el accidente. También se confirmó experimentalmente la posibilidad de estimar la presión de ruptura de la vaina del combustible, a partir de la resistencia mecánica de los materiales. Además se observó un alargamiento menor después de la ruptura en el tubo de Zircaloy-2 con un contenido de hidrógeno absorbido de alrededor de 300 ppm, aunque se encontró que su resistencia mecánica habia cambiado poco.

Etude expérimentale de l'inversion du débit dans le réacteur belge BR2*

par G. Stiennon, K. J. Bobin, H. Dopchie, L. Fouarge, F. Leonard, F. Mathieu, F. Motte, J. Planquart, J. Ransbotyn, D. Stoian et R. Vanhaelewyn **

Le réacteur BR2 est un réacteur hétérogène à neutrons thermiques ayant comme modérateurs l'eau légère et le béryllium, et comme réfrigérant l'eau légère.

L'élément combustible est formé de six plaques concentriques (fig. 1), le combustible est de l'uranium 235 enrichi à 90%, la partie combustible de la plaque est un alliage Al-U, le gainage est en aluminium.

Les principales dimensions d'une plaque combustible sont:

longueur de la partie active: 762 mm
épaisseur de la partie active: 0,5 mm
épaisseur totale: 1,27 mm
longueur totale: 965 mm.

L'eau légère servant comme modérateur et réfrigérant circule normalement de haut en bas. Le schéma du circuit primaire de réfrigération est donné sur la figure 3. Le régime nominal d'écoulement de l'eau est:

pression à l'entrée: 13,6 atm (abs.)
pression à la sortie: 10,1 atm (abs.)
vitesse moyenne dans l'élément combustible: 11 m/s
température de l'eau à l'entrée: de 30 à 40 °C.

DÉFINITION GÉNÉRALE DES ESSAIS

L'objectif des essais de montée en puissance du réacteur BR2 était d'une part, de montrer que le réacteur peut fonctionner en sécurité au flux de chaleur maximal de 400 W/cm^2, valeur pour laquelle il a été prévu, d'autre part, de rechercher jusqu'à quel flux de chaleur maximal la sécurité de fonctionnement est conservée.

L'ensemble des essais de montée en puissance a consisté en une suite d'étapes de fonctionnement du réacteur à une puissance totale constante. Pendant chaque étape, une perturbation entraînant un manque de débit de refroidissement primaire est provoquée, et la réponse du complexe est enregistrée.

* Travail exécuté dans le cadre de l'Association CEN-EURATOM chargée d'exploiter le réacteur BR2 et ses installations connexes (contrat 006/60/5 BRAB).

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ORGANISATION ET INSTRUMENTATION DES ESSAIS

Système de mesure et de commande

Le noyau (cœur) du réacteur utilisé pour ces essais se compose de 14 éléments de combustible Sylcor à 6 tubes chacun, 7 barres de contrôle et 2 barres de régulation.

Les canaux occupés par ces ensembles sont donnés à la figure 2. Les autres canaux de la matrice sont occupés par des bouchons béryllium standard.

Parmi les ensembles combustibles, quatre sont munis de prise de pression et de thermocouples. Ils sont chargés dans les canaux GO, F346, C41, H323. Pour chacun de ces canaux, ces ensembles instrumentés permettent de mesurer:

a) La différence de pression aux bornes de l'élément combustible et donc après calibration, le débit d'eau dans le canal;

b) La température de l'eau traversant le canal, température mesurée de part et d'autre de l'élément combustible;

c) La température de l'eau en haut du passage VI de l'élément (fig. 1);

d) La température de paroi le long d'une génératrice extérieure du tube 6 à 5 niveaux régulièrement espacés de 150 mm et disposés symétriquement par rapport au plan médian (fig. 1).

Pour la température de paroi, les thermocouples sont sertis puis soudés par ultra-sons dans la gaine extérieure du tube 6. La soudure chaude se trouve au milieu d'un des trois secteurs constituant le tube 6 (fig. 1).

Les températures rapidement variables lors d'une perturbation sont relevées sur des enregistreurs à déroulement rapide (3 mm/s), le déroulement étant déclenché par le système de commande de l'incident.

Les chaînes de mesure du système de contrôle du réacteur permettent la mesure du bilan thermique du réacteur ainsi que de l'activité ^{16}N en un point du circuit primaire.

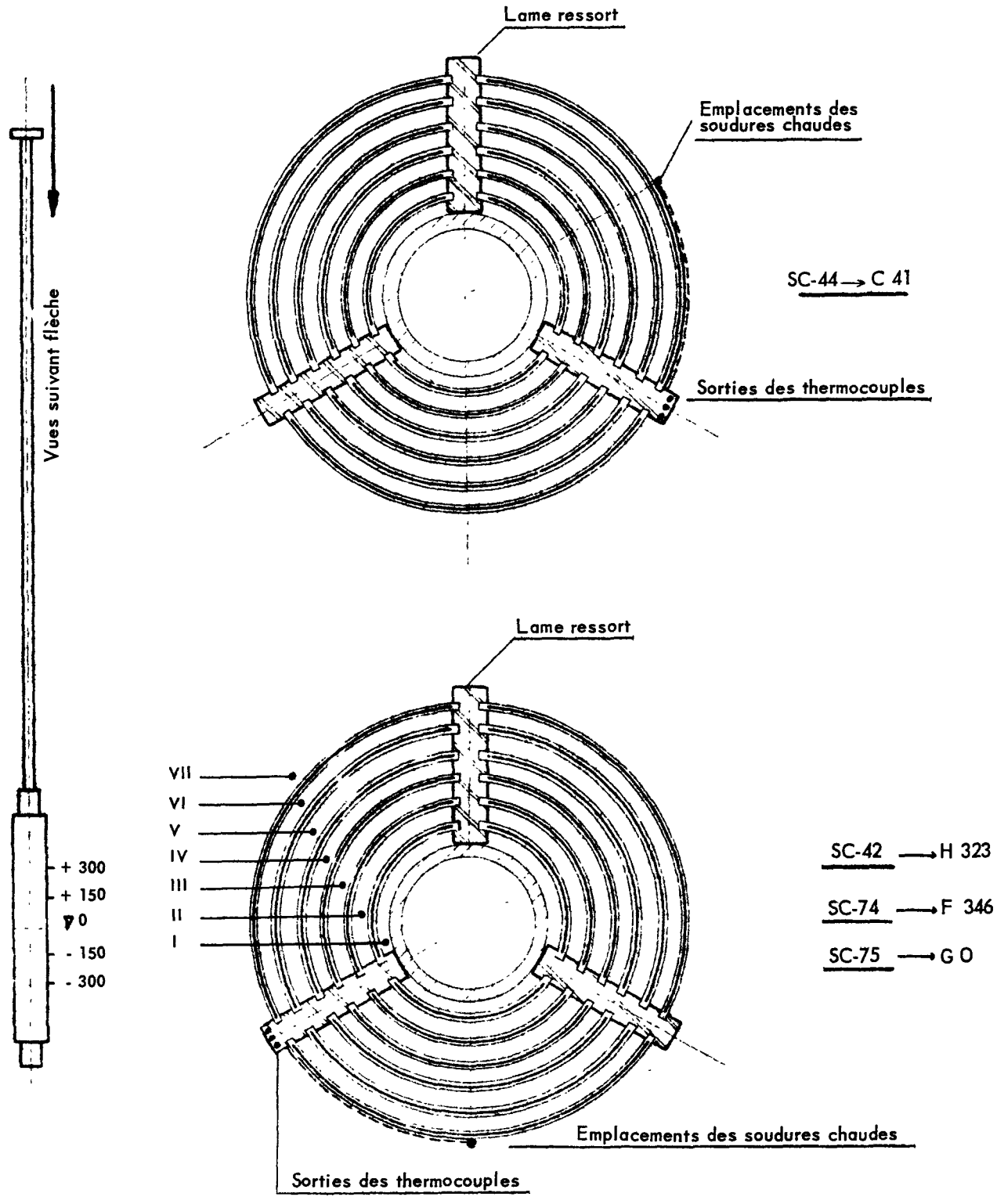


Figure 1. Emplacements des soudures chaudes des thermocouples des éléments instrumentés

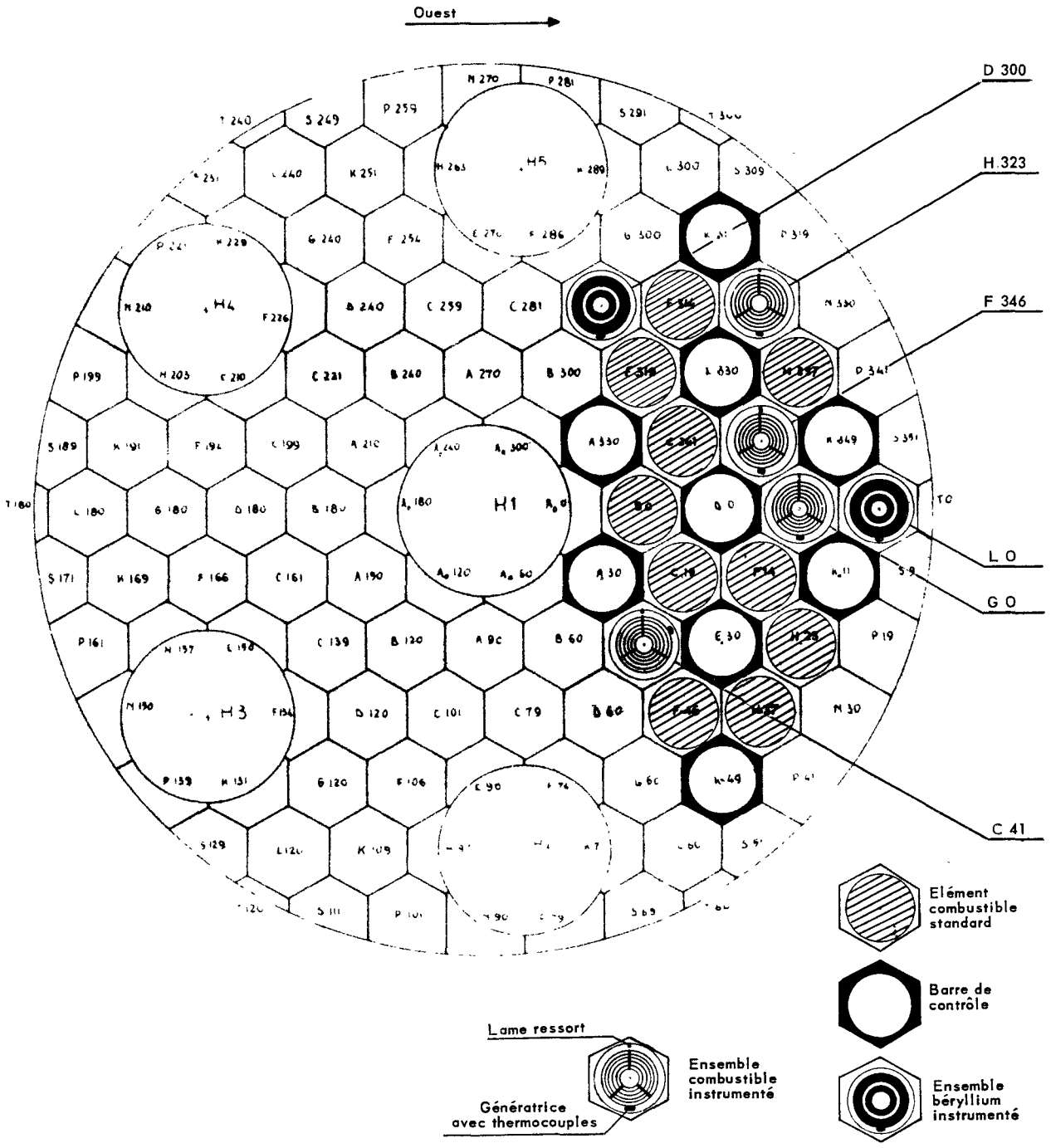


Figure 2. Noyau de démarrage BR2

La séquence des événements importants survenant à la suite du déclenchement d'une perturbation est obtenue par l'impression sur un enregistreur Visicorder d'une série de signaux prélevés en différents points du complexe.

En vue d'obtenir une précision suffisante dans l'enregistrement de la réponse du complexe à une perturbation, le système de commande permet de déclencher toute perturbation à partir d'un seul bouton-poussoir pour autant que toutes les conditions requises soient réalisées.

Etalonnage et vérification

Les différences de pression aux bornes des éléments combustibles ont été calibrées en débit dans une boucle hydraulique reproduisant un canal standard du réacteur.

Après le chargement des ensembles instrumentés dans le réacteur, la température de l'eau du circuit primaire a été augmentée de 20 à 40 °C au moyen de l'énergie dissipée dans l'eau par les pompes principales. Les lectures du système de mesure auxiliaire et des

thermorésistances standard ont permis de s'assurer que chaque ligne de mesure fonctionnait et que l'allure de l'ensemble des courbes était cohérente.

Avec le réacteur à l'arrêt, le système de commande a été vérifié en exécutant successivement les diverses perturbations et en examinant les enregistrements obtenus pour constater que les valeurs des grandeurs observées étaient conformes aux valeurs prévues.

Au cours des essais, des variations non prévues de lecture des thermocouples sont apparues et des calibrations supplémentaires ont dû être exécutées.

A la fin des essais, l'examen des ensembles instrumentés a montré des défauts d'étanchéité dans les connexions des thermocouples. Ces défauts peuvent expliquer les dérives constatées, de sorte que l'analyse des lectures de calibration exécutées pendant ces essais ne peut fournir des informations utiles concernant l'effet des rayonnements sur les thermocouples utilisés.

Procédure générale des essais

L'analyse du fonctionnement du circuit primaire (fig. 3) avec une commande automatique ou manuelle de la vanne de by-pass (V4) indique qu'une réduction du débit de refroidissement du noyau résulte de l'une des perturbations suivantes:

Repère *Définition*

A : Perte totale de débit suivie d'ouverture de la vanne de by-pass V4;

B : Perte totale de débit sans ouverture de la vanne de by-pass V4;

C : Ouverture intempestive de la vanne de by-pass V4;

D : Fermeture intempestive des vannes de blocage V1, V2 suivie d'ouverture de la vanne de by-pass V4;

E : Fermeture intempestive des vannes de blocage V1, V2 sans ouverture de la vanne de by-pass V4;

F : Perte totale de pression suivie d'ouverture de la vanne de by-pass V4;

G : Perte totale de pression sans ouverture de la vanne de by-pass V4;

H : Transfert automatique des pompes primaires principales;

O : Perte du débit principal par manque de tension 10 kV simulé;

Le programme a commencé par une série de perturbations Rep C exécutées sur un noyau pratiquement frais à des flux de chaleur maximaux croissant à partir de 200 W/cm² par palier de 100 W/cm² jusqu'à la valeur maximale de 600 W/cm².

Les autres perturbations ont été exécutées à 400 W/cm².

De manière à préciser la réponse du complexe pendant le cycle de fonctionnement normal en même temps qu'aux environs du flux de chaleur maximal prévu (400 W/cm²), certaines de ces perturbations ont été exécutées à des flux de chaleur maximaux de 300, 500 et 600 W/cm² et à des épuisements croissants du combustible.

FONCTIONNEMENT A PUISSANCE TOTALE CONSTANTE

Mesure de la puissance

Puissance dégagée dans un canal

A partir des mesures fournies par l'ensemble combustible instrumenté, la puissance $N\,(i)$ dégagée dans un canal se calcule d'après les formules habituelles.

L'analyse des erreurs donne, pour les puissances totales du réacteur supérieures à 24 MW, une erreur sur la puissance dégagée par canal égale à 13 %. Dans cette valeur, 2 % sont dus à l'erreur sur le débit et 11 % à l'erreur sur la différence de température.

Puissance totale dégagée dans le réacteur

Pendant les essais de montée en puissance, le bilan thermique du circuit primaire et du circuit piscine a été établi régulièrement en même temps qu'était mesurée l'activité ^{16}N du circuit primaire à la sortie du réacteur.

En supposant que la puissance dégagée dans un canal est proportionnelle au flux moyen de neutrons thermiques dans ce canal, les mesures neutroniques faites sur le noyau de démarrage à BR02, barres de contrôle à 480 mm, permettent de calculer la puissance totale $\Sigma\, N\,(i)$: somme des puissances dégagées dans les différents canaux. BR02 est le modèle nucléaire de BR2. La cote 0 mm pour les barres de contrôle correspond à l'état: barres de contrôle complètement introduites dans le noyau.

Cette puissance $\Sigma\, N\,(i)$ peut être comparée à la puissance totale N_{tot} mesurée par l'activité N 16, au moyen du coefficient

$$\delta = \frac{N_{tot} - \Sigma\, N\,(i)}{N_{tot}} \cdot 100\ [\%]$$

Le tableau 1 donne les valeurs obtenues pour cinq régimes de fonctionnement à puissance totale variant

Tableau 1. Puissance thermique et puissance nucléaire

	1	3	4	5	6
N (F346) (MW)	0,97	1,38	1,94	2,45	2,87
N (GO) (MW)	0,90	1,38	1,74	2,27	2,73
N (C41) (MW)	0,74	1,08	1,51	1,93	2,29
N (H323) (MW)	0,60	0,84	1,30	1,68	1,95
$\Sigma\, N\,(i)$ (MW)	11,09	16,02	22,54	28,88	34,02
I_{Ai} (5000) (10^{-8} A)	2,8	4,1	5,5	7,1	8,4
N_{tot} (MW)	12,15	17,80	23,85	30,80	36,45
δ (%)	8,7	10,1	5,5	6,2	6,6

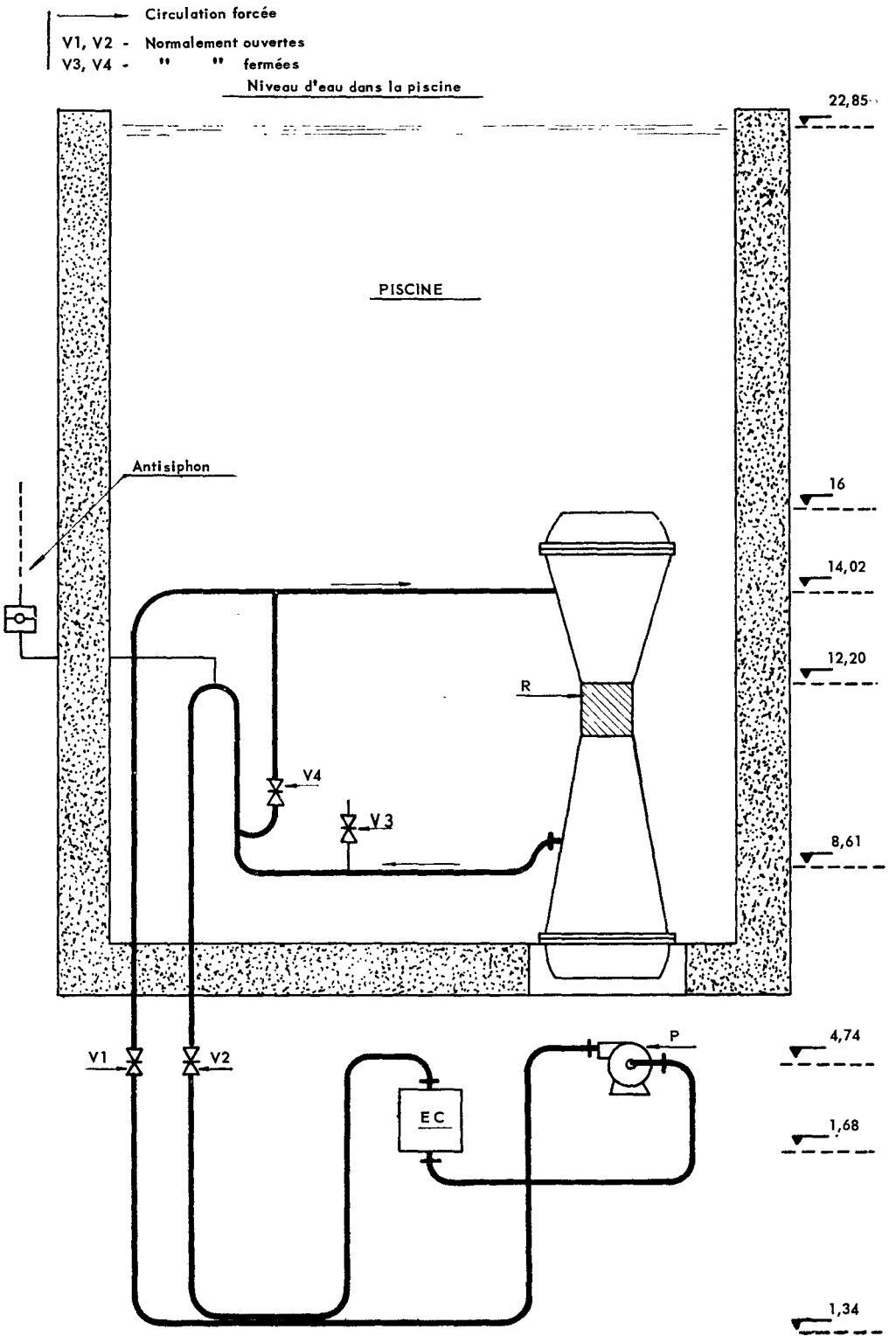


Figure 3. Schéma du circuit primaire de réfrigération.

R – noyau du réacteur, P – pompes du primaire, V – vannes automatiques, EC – échangeurs de chaleur, ▼ niveau en mètres

entre 12 et 36 MW. Pour ces cinq régimes, la position des barres de contrôle a varié entre 490 et 505 mm.

L'écart entre les deux valeurs de la puissance totale est systématique: la puissance totale obtenue par bilan thermique est supérieure à celle obtenue par sommation des puissances dégagées dans les canaux chargés en combustible, ce qui est conforme aux prévisions.

L'erreur sur N_{tot} peut être considérée égale à 3,5%. En acceptant une erreur de 13% sur la puissance dégagée par canal et une erreur de 1% sur les coefficients de répartition de puissance par canal, la précision de $\Sigma N(i)$ est égale à 3,5%.

En conséquence, le coefficient δ mesure une différence significative et on peut considérer que le bilan thermique mesuré par l'activité ^{16}N fournit une puissance totale dégagée par le réacteur supérieure de 6% à la puissance totale dégagée dans les canaux chargés d'éléments combustibles standard.

Le tableau 2 indique que la distribution des puissances dégagées par canal chargé d'éléments combustibles standard est conforme à la distribution des flux moyens de neutrons thermiques par canal tels que mesurés à BR02.

Tableau 2. Rapport des distributions des puissances et des flux neutroniques

	1	3	4	5	6	BR02
F346	1,22	1,21	1,20	1,19	1,18	1,18
GO	1,14	1,21	1,08	1,10	1,12	1,10
C41	0,93	0,94	0,94	0,94	0,94	0,97
H323 . . .	0,76	0,73	0,81	0,82	0,80	0,79

Mesure de la température de paroi des plaques combustibles

Flux de chaleur

A partir des mesures faites sur BR2 et sur BR02, le flux de chaleur maximal nominal dans un noyau à n éléments peut se définir par la relation suivante:

$$Q_{\max\ nom} = \frac{N_{tot}}{n\,A\,K} \cdot f_{macro} \cdot f_{axial} \cdot f_{micro}$$

A: Surface totale d'échange d'un élément combustible = $1{,}258 \times 10^4$ cm²,

K: Rapport entre la chaleur dégagée dans les plaques et celle emportée par l'eau de refroidissement = 1,055,

$$f_{macro} = \frac{\Phi_{moy\ canal}}{\Phi_{moy\ noyau}} = 1{,}21$$

$$f_{axial} = \frac{\Phi_{max\ canal}}{\Phi_{moy\ canal}} = 1{,}85$$

$$f_{micro} = \frac{\Phi_{max\ local}}{\Phi_{max\ canal}} = 1{,}28$$

Cette relation ne tient pas encore compte du coefficient défini par le tableau 1.

Les cartes de flux de neutrons thermiques établies à BR02 indiquent que, avec les barres de contrôle entre 485 et 505 mm, le flux de chaleur maximal nominal se trouve dans le passage VI cu canal C 341 à 150 mm en dessous du plan médian.

Sur les éléments combustibles instrumentés, les thermocouples les plus proches du point chaud sont placés sur le tube 6 à 150 mm en dessous du plan médian dans les canaux F 346 (TC14) et GO (TC24). Avec les coefficients fournis par ces mêmes cartes, on établit les relations suivantes:

$$q\ (TC14) = 0{,}91\ q_{\max\ nom} \qquad q\ (TC24) = 0{,}80\ q_{\max\ nom}$$

Le tableau 3 donne les flux de chaleur maximaux nominaux obtenus pendant les essais.

Tableau 3. Flux de chaleur (W/cm²)

	1	3	4	5	6
$q_{\max\ nom}$ . .	188	276	370	477	565

En acceptant une erreur de 3,5% sur N_{tot} et de 2% sur chacun des 3 coefficients f, on obtient une erreur de 9,5% soit 10% sur le flux de chaleur maximal nominal.

Coefficients de transfert de chaleur aux endroits des thermocouples

Pour les différents régimes de fonctionnement, nous disposons des valeurs des températures indiquées par les thermocouples TC 14 et TC 24, des valeurs de l'augmentation de température de l'eau dans les canaux F 346 et GO et des valeurs des flux de chaleur à l'emplacement des thermocouples TC 14 et TC 24.

A partir de ces valeurs, des cartes de flux de neutrons thermiques et des caractéristiques hydrauliques et mécaniques des éléments combustibles, il est possible de calculer la température de paroi et la température de l'eau au droit des thermocouples et de là, le coefficient de transfert de chaleur h entre la paroi et l'eau en ces mêmes points.

Température maximale nominale

D'après ce qui précède, le point (Z_m) du flux de chaleur maximal nominal est situé dans le passage 6 du canal C341 à 150 mm en dessous du plan médian.

A partir de la mesure de ΔT dans le canal F346 on peut calculer la température nominale de l'eau au point Z_m.

Connaissant la température de l'eau, le flux de chaleur et la courbe $h_6 = f(T_m)$ au point Z_m, il est possible de déterminer au moyen de quelques itérations

la température de la paroi au point Z_m. Ces itérations se font sur la formule:

$$T_w(Z_m) = T_{eau}(Z_m) + \frac{q_{max\ nom}}{h_6}$$

Erreurs de mesure et tolérances constructives

Pour établir une limite de fonctionnement, nous devons considérer les erreurs de mesure en même temps que les tolérances constructives des éléments combustibles qui influencent la température de paroi.

Pour déterminer la température maximale probable nous considérons que tous les facteurs défavorables (entraînant une augmentation de la température de paroi) apparaissent au point chaud suivant une distribution statistique normale et nous prenons la valeur extrême probable au point Z_m pour les trois grandeurs suivantes: température de l'eau, flux de chaleur maximal, coefficient de transfert de chaleur.

Ces valeurs sont fournies par les relations suivantes:

$$\text{prob. } T_{eau}(Z) = T_{in} + 0{,}5 + 1{,}082\,\Delta T$$
$$q_{max\ prob} = 1{,}27\ q_{max\ nom}$$
$$h_{min\ prob}(Z_m) = 0{,}88\ h(Z_m).$$

En itérant comme dans le paragraphe précédant, il est possible de calculer $T_{max\ prob}$ en fonction du flux de chaleur maximal nominal.

Pour déterminer la température maximale possible, nous considérons que tous les facteurs défavorables apparaissent au point chaud simultanément et nous prenons les valeurs extrêmes possibles pour les trois grandeurs permettant le calcul de $T_w(Z_m)$.

Comparaison des profils de température et des flux neutroniques, valeurs mesurées

La figure 4 donne pour le canal GO le profil des différences de température entre la paroi et l'eau pour les différents régimes de fonctionnement et pour le canal BO qui lui est symétrique, le profil des flux neutroniques mesurés dans l'axe du canal.

On observe une bonne concordance entre ces profils, ce qui confirme la faible dépendance du coefficient de transfert vis-à-vis de la température de film dans la plage mesurée. Ces mêmes profils ont été tracés pour les autres canaux instrumentés; la même concordance est observée. De plus, ces profils indiquent que les 19 thermocouples utilisés se sont trouvés en des points où les conditions de fonctionnement sont les conditions nominales.

Limite de fonctionnement en régime stable

Température admissible

La température admissible est définie par la relation

$$T_{adm} = T_{sat} + S$$

T_{sat} = température de saturation de l'eau à la pression existant au point chaud;

S = surchauffe admise.

Les conditions nominales de fonctionnement du circuit primaire donnent $T_{sat} = 185$ °C. En admettant $S = 20$ °C on obtient $T_{adm} = 205$ °C.

Critères possibles

Sur la figure 5 sont représentées en fonction du flux de chaleur maximal ou de l'activité ^{16}N les différentes températures déduites des observations faites pendant les cinq régimes de fonctionnement analysés précédemment.

Pour déterminer la puissance totale permise sans mesurer la température au point chaud, on peut utiliser l'un des critères suivants:

1) $T_{max\ poss} \leq T_{adm}$ 2) $T_{max\ prob} \leq T_{adm}$
3) $T_{max\ nom} \leq T_{adm}$

Avec le critère 1, le flux de chaleur maximal nominal permis est 490 W/cm². Dans ces conditions, l'ébullition locale est pratiquement exclue.

Pendant les essais, BR2 a fonctionné à 565 W/cm² pendant 10 h sans que l'on observe avec l'appareillage existant une ébullition locale ou une modification de la stabilité de fonctionnement de la chaîne de régulation. L'activité produits de fission n'a pas été modifiée de manière significative, indiquant qu'il n'y a pas eu de ruptures de gaine. En conséquence, le critère 2 peut être considéré comme suffisant. Il donne un flux de chaleur maximal nominal permis de 600 W/cm².

Pour pouvoir utiliser le critère 3, il faut encore étudier l'influence d'une ébullition locale sur la stabilité de fonctionnement du réacteur et sur la corrosion des plaques. Si ce critère est acceptable, le flux de chaleur maximal nominal permis est de l'ordre de 800 W/cm².

RÉPONSE AUX PERTURBATIONS

Evénements observés

A chaque type de perturbation correspond un déroulement déterminé des événements suivants:

a) Largage par relais des barres de contrôle;

b) Mouvement des vannes V1, V2, V3, V4 (fig. 3);

c) Arrêt des pompes principales suivi de la fermeture des clapets.

Ce déroulement est indépendant de la puissance de fonctionnement du réacteur et son enregistrement peut être considéré comme une mesure de la stabilité de fonctionnement de l'ensemble de ces organes de sécurité.

Lors de chaque perturbation, on enregistre pendant 3 min:

a) Les températures de paroi des ensembles combustibles instrumentés;

b) La température de l'eau en haut du passage VI;

c) La température de l'eau en dessous de l'élément combustible.

La figure 6 donne une réponse typique: celle après une perturbation de type C exécutée après fonctionnement du réacteur à une puissance totale de 36 MW, soit $q_{\max\ nom} \leq 560$ W/cm². Sur cette figure $t = 0$ correspond au repère d'enregistrement inscrit par un signal envoyé par le poste de commande.

La figure 7 donne pour les différents types de perturbations les températures maximales de paroi mesurées après la perturbation en fonction de la puissance totale avant la perturbation. Ces températures sont mesurées dans le canal F346 au thermocouple 14. Pour chaque type de perturbation, ces mesures sont faites sur un noyau de faible épuisement. Avec des épuisements croissants, la lecture sur TC 14 a tendance à diminuer tandis que les lectures sur TC 13 et TC 12 placés au dessus de TC 14 respectivement à 150 et 300 mm aug-

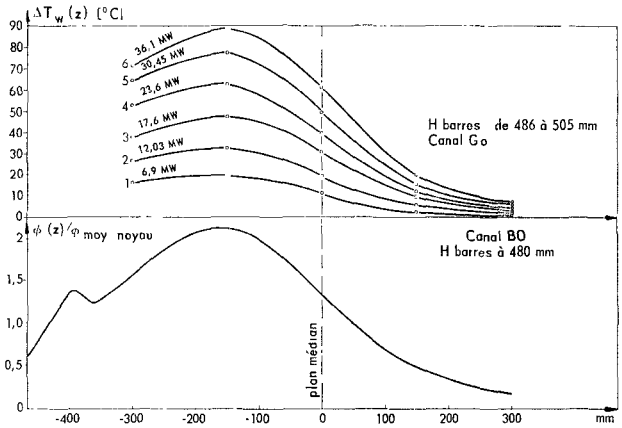


Figure 4. Profils parallèles à l'axe du canal indiqué

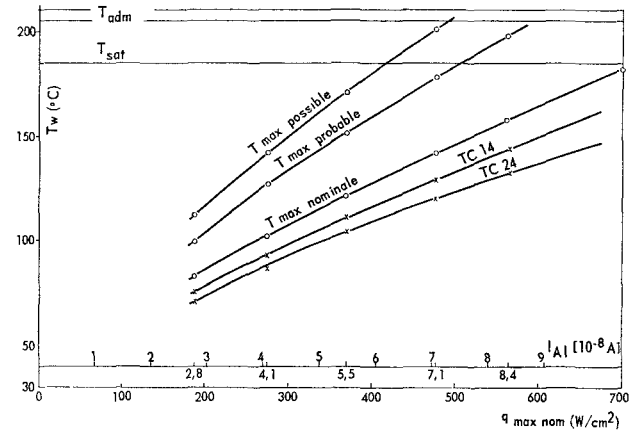


Figure 5. Température de paroi en fonction de la puissance totale: ⊙ valeurs calculées, X valeurs mesurées

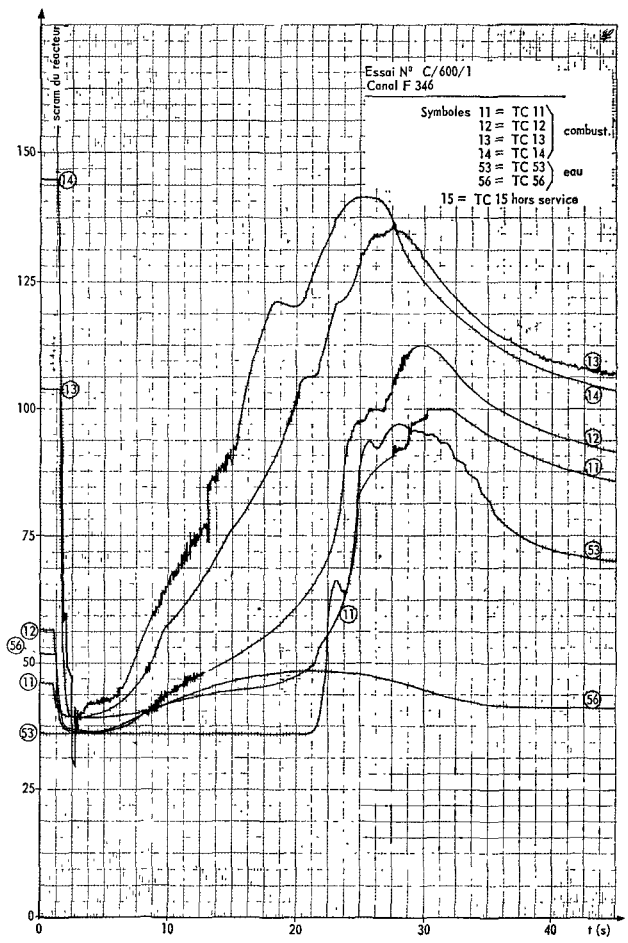


Figure 6. Température de paroi et d'eau en fonction du temps après perturbation

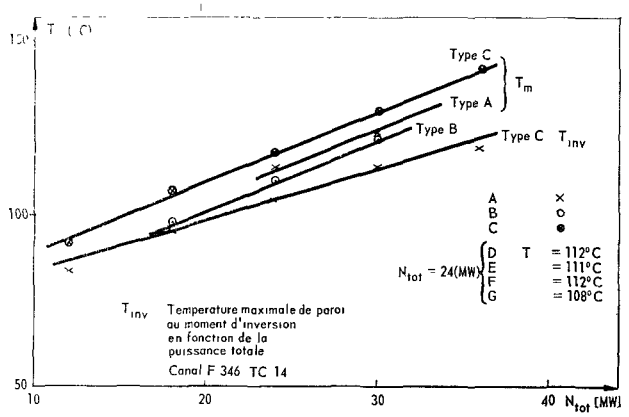


Figure 7. Température maximale de paroi mesurée après perturbation en fonction de la puissance totale

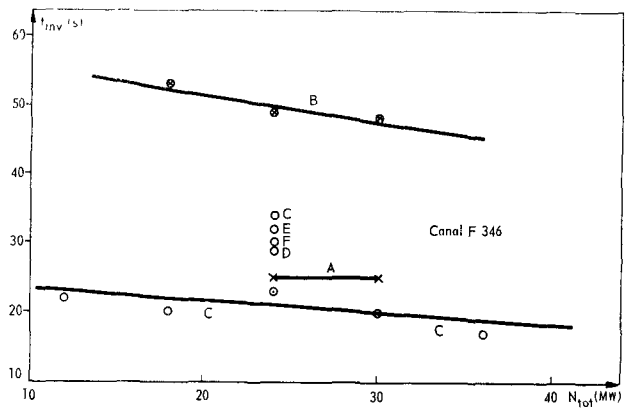


Figure 8. Moment de l'inversion du débit en fonction de la puissance totale

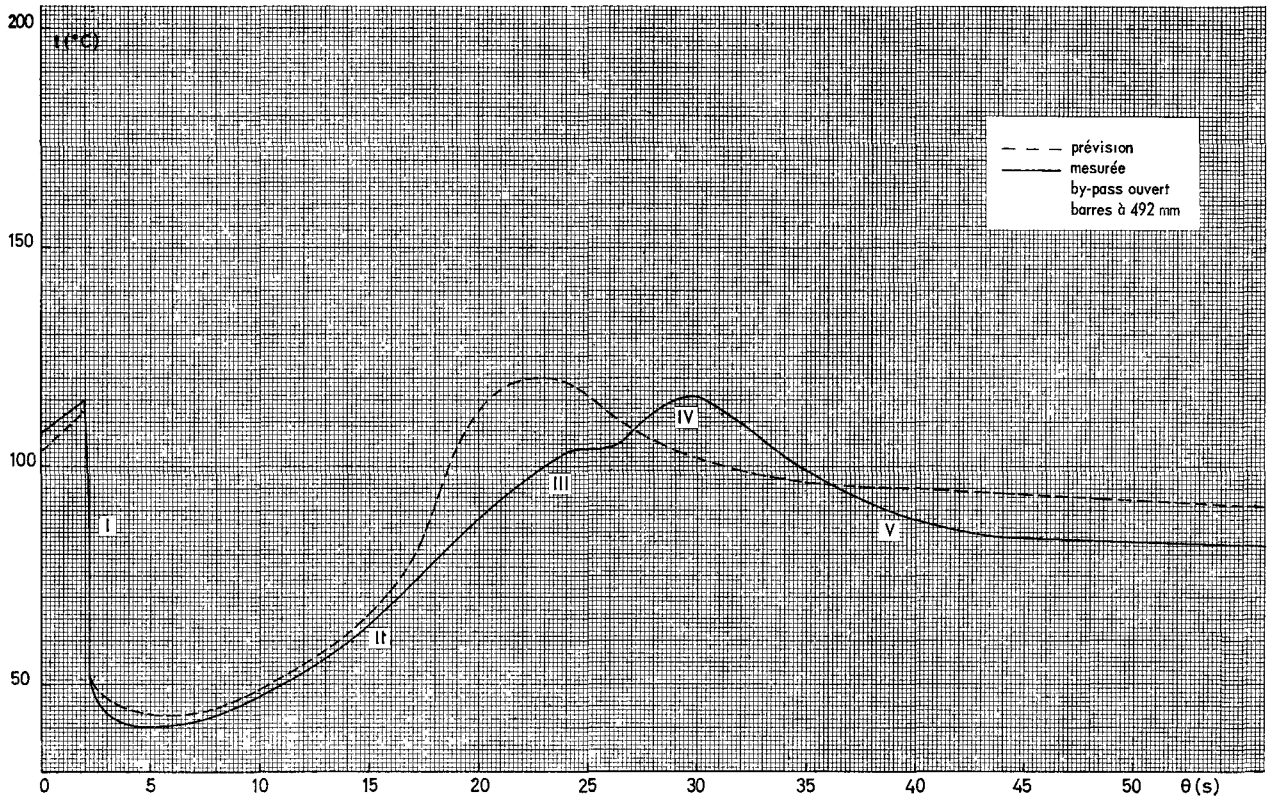


Figure 9. Température maximale de la paroi au niveau 15 cm pendant les transitoires

mentent jusqu'à se confondre avec TC 14 à quelques degrés près.

Inversion de débit

La circulation forcée de l'eau de refroidissement se fait de haut en bas dans les éléments combustibles. La circulation naturelle qui s'établit après une perte totale de débit entraîne un renversement du sens de circulation de l'eau dans les éléments combustibles. Le moment où la vitesse d'eau passe par zéro est appelé le moment d'inversion (t_{inv}). Le moment $t = 0$ correspond au moment de la chute de température de plaque lors du largage des barres de contrôle.

Pour déterminer ce moment d'inversion, nous utilisons l'enregistrement de la température d'eau indiquée par le thermocouple placé en haut du passage VI.

Le calcul de t_{inv} est basé sur les deux hypothèses suivantes:

a) Après le moment d'inversion, la vitesse ascensionnelle de l'eau augmente linéairement, le mouvement a donc une accélération constante a;

b) La valeur maximale de la température est celle de l'eau qui au moment t_{inv} se trouve au bord inférieur de la partie active de la plaque combustible.

La figure 8 donne le moment d'inversion pour le canal F346 en fonction de la puissance totale du réacteur pour les différents types de perturbations.

Pour ces mêmes perturbations, les moments d'inversion des canaux instrumentés diffèrent au maximum de 4 s et en moyenne de 2 s. L'erreur estimée du moment d'inversion est $\pm$ 2 s.

La figure 7 donne pour le canal F346, la température maximale de paroi mesurée au moment d'inversion en fonction de la puissance totale du réacteur pour les perturbations de type C. Cette température est indiquée par le thermocouple TC 14.

La figure 9 permet de comparer, pour un essai à 400 W/cm^2, les résultats expérimentaux et les prévisions théoriques établies à partir des formules habituelles utilisées pour l'étude des échanges calorifiques.

Conclusions

Les conclusions ne peuvent s'appliquer qu'à un noyau de même configuration que celle du noyau de démarrage et constitué d'éléments combustibles de même type que les éléments ayant constitué le noyau de démarrage:

a) La circulation naturelle s'établit en même temps dans les différents canaux du noyau.

b) Au point de vue des températures de paroi atteintes après perturbation, les perturbations de type C sont les plus sévères. Il est donc nécessaire d'examiner si l'automaticité de la vanne de by-pass doit être maintenue.

c) Au point de vue des risques d'ébullition, les perturbations de type F ou G (perte de pression) sont les plus sévères parce qu'elles réduisent la température admissible. Pour une puissance totale de 24 MW, la température maximale atteinte dans le canal F346 est 112 °C et la température maximale au moment de l'inversion est 98 °C. Ces valeurs sont inférieures aux valeurs limites fixées par l'analyse de sécurité des essais (respectivement 120 et 100 °C).

d) La confiance acquise dans les méthodes de calcul utilisées permet d'en envisager l'application à d'autres configurations.

CONCLUSIONS GÉNÉRALES

Le réacteur BR2 est capable de fonctionner en sécurité à un flux de chaleur maximal nominal de 400 W/cm² donc à la valeur pour laquelle il a été conçu.

A ce flux de chaleur, correspondent:

a) Un flux de neutrons thermiques maximal sur l'axe du canal le plus chaud égal à $6{,}0.10^{14}$ n/cm² s ($\Phi_{th} = \int_0^{0{,}5\ eV} \Phi\,(E)\,dE$)

b) Un flux de neutrons rapides ($E > 1$ MeV) au même point égal à $3{,}0.10^{14}$ n/cm² s.

Pour permettre le fonctionnement à 600 W/cm², il faut calculer les risques résultant d'une perte de pression du circuit primaire.

Pour permettre le fonctionnement à 800 W/cm² il faut calculer les risques résultant d'une ébullition locale existant normalement pendant le fonctionnement à puissance stable en même temps que les risques résultant d'une ébullition généralisée se produisant après une perturbation entraînant une réduction du débit de refroidissement primaire.

ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/517 Belgium

Experimental study of flow inversion in the Belgian engineering test reactor BR2

By G. Stiennon *et al.*

After briefly reviewing the main features of BR2, the Belgian high-flux engineering test reactor, the authors go on to describe a series of tests carried out recently to determine its maximum safe operating power. The tests consisted essentially in measuring the temperature of the surface of the fuel plates by means of thermo-couples embedded in the cladding, at different reactor powers, while deliberately provoking all possible breakdown conditions; the latter included total loss of coolant flow (by shutting down the pumps, by opening a by-pass valve in the core or by closing the valves in the primary circuit) and loss of pressure in the primary circuit. Simultaneously, the flux was measured in the core. The organization of the tests and the equipment used are described in the paper.

During the tests, the maximum power reached was 37 MW, giving a maximum heat flux of 600 W/cm² at the fuel-element surface. This is 1.5 times the designed figure, and is exceptionally high for the type of fuel element in question.

The corresponding thermal neutron flux (energy less than 0.5 eV) in the fuel plates was 10^{15} n/cm² s. Even under these extreme conditions, the sudden loss of coolant followed by flow inversion in the core did not lead to impermissibly high plate temperatures, provided the primary pressure was maintained.

The results obtained, both at steady power and during the different tests, and in particular for the most severe transient, are analysed.

The time taken to reach the moment of inversion having been determined, the maximum "hot-spot" temperatures in the fuel are calculated. The upper limit of power density to ensure that boiling does not occur even in the hottest channel is then determined.

These data will make it possible to assess the safety of this type of reactor at high power density and for different configurations of the core.

A/517 Бельгия

Экспериментальное изучение изменения расхода теплоносителя в бельгийском реакторе BR-2

Г. Стьеннон *et al.*

После краткого описания основных характеристик реактора для испытаний материалов BR-2 с высоким потоком нейтронов авторы описывают результаты испытаний, которые были недавно проведены с целью определения мощности, при которой обеспечивается безопасная эксплуатация реактора. Эти испытания касались главным образом измерения температуры тепловыделяющих элементов с помощью термопар, установленных в оболочках, для различной мощности реактора, причем намеренно создавались аварийные условия эксплуатации реактора, такие, как потеря расхода водяного теплоносителя (путем остановки насосов, путем открывания контура теплоносителя в обход ак-

тивной зоны реактора, закрывания клапанов, расположенных в первом контуре) и потеря первичного давления. Параллельно проводились нейтронные измерения в активной зоне реактора. Описывается постановка испытаний и оборудование, необходимое для их проведения.

Во время испытаний максимальная мощность, достигаемая реактором, составила 37 *Мвт;* она соответствует максимальной удельной мощности 600 *вт/см*^{2} на поверхности уранового тепловыделяющего элемента, что в полтора раза превышает номинальную удельную мощность и является исключительной для тепловыделяющих элементов используемого типа. Действительный поток тепловых нейтронов (с энергией менее 0,5 *эв*) в урановых пластинах составил 10^{15} *нейтр/см*$^2 \cdot$ *сек.* Даже при этих предельных условиях (причем поддерживалось первичное давление) резкое прекращение охлаждения, сопровождаемое изменением расхода воды в активной зоне, не создавало слишком высокой температуры в пластинах.

В докладе приводится анализ результатов для работы на постоянной мощности и для различных нарушений, в частности для наиболее опасных переходных режимов мощности.

После определения запаздывания изменения расхода теплоносителя выводятся температуры, достигаемые топливными пластинами в наиболее горячем месте. Затем определяется предельная удельная мощность, при которой в наиболее горячем канале вода не закипает.

Эти данные позволят уточнить безопасность работы реактора такого типа при высокой удельной мощности и для различных конфигураций активной зоны.

A/517 Bélgica

Estudio experimental de la inversión del flujo en el reactor belga BR2

por G. Stiennon *et al.*

Después de describir brevemente las características generales del reactor de alto flujo BR2 para ensayo de materiales, los autores informan sobre los ensayos efectuados recientemente para determinar hasta qué potencia se puede utilizar el reactor en condiciones de seguridad. Dichos ensayos se han referido esencialmente a la determinación de la temperatura de las placas combustibles, mediante termopares introducidos en las vainas, para diferentes potencias del reactor mientras se producía deliberadamente accidentes tales como pérdidas de agua de refrigeración (por parada de las bombas, por apertura de un circuito en derivación con el reactor, por cierre de las válvulas del circuito primario) y pérdida de la presión. Al mismo tiempo se efectuaban mediciones neutrónicas en el núcleo del reactor. La memoria describe la organización de los ensayos y el equipo necesario.

La potencia máxima alcanzada por el reactor durante los ensayos fue de 37 MW, que corresponde a una potencia específica máxima de 600 W/cm^2 en la superficie de los elementos de uranio, es decir, un valor 1,5 veces superior a la potencia específica nominal y un valor excepcional para el tipo de elementos empleados. El flujo real de neutrones térmicos (de energía inferior a 0,5 eV) alcanzado en las placas de uranio fue de 10^{15} n/cm^2 s. Incluso en estas condiciones extremas, si se mantiene la presión, la interrupción súbita de la refrigeración seguida de la inversión del paso del agua en el núcleo no ocasiona temperaturas excesivamente elevadas en las placas.

La memoria presenta el análisis de los resultados para funcionamiento a potencia constante y para los diversos accidentes, en particular para el estado transitorio más peligroso.

Una vez determinado el retardo en la inversión del paso de agua, se deducen las temperaturas alcanzadas por las placas combustibles en el punto más caliente. La potencia específica límite viene entonces determinada por la condición de que no hierva el agua en el canal más caliente.

Estos datos permitirán apreciar la seguridad de este tipo de reactor a potencias específicas elevadas y para diversas configuraciones del núcleo.

Design of an electrostatic precipitator monitoring system for ruptured fuel element detection

By B. S. Prabhakar, R. Sethuraman, K. Srinivasan and K. Venkatachalam *

A system for detecting rupture in the fuel cladding is a common and a necessary feature of all heterogeneous reactors, research and power alike. However, there is a wide range in the type and extent of the system used for the purpose of detecting ruptured fuel elements [1]. The system is largely influenced by the type of cladding used, its susceptibility to rupture and the extent of fouling that may result from a fuel rupture. At one end of the range are the Calder Hall type of power reactors which have Magnox clad, natural uranium fuel elements working at the limit of the permissible temperatures; these reactors employ a complex system—monitoring groups of fuel channels continuously and using large data processing equipment. At the other end are the light-water power reactor which use stainless steel clad, enriched uranium fuel rods; these reactors employ continuous monitoring of just the gamma activity of the bulk coolant off-gas, and individual channel location is done off load.

The performance of a ruptured fuel detection system can be defined by three main characteristics:

(*a*) *The sensitivity*. This can be defined as the minimum area of rupture of the can which can be monitored by the system. The sensitivity is very much dependent on the signal-to-background discrimination. Data storage and analysis methods are important since the background level is dependent on the operational history of the reactor as a whole and the individual fuel channels.

(*b*) *The time delay in monitoring*. This is determined by the delay which can be tolerated between a fuel rupture and its detection. Generally this delay has also a bearing on the sensitivity of the rupture detection system. Thus, a system can be used to monitor individual fuel elements (or small groups of fuel channels) cyclically instead of monitoring the reactor as a whole thus reducing the dilution of the activity due to a fuel rupture. A larger acceptable time delay may also result in a lower investment on coolant sample pumping systems.

(*c*) *The faulty element location capability*. Some systems are capable of locating the faulty fuel element directly or with a little manipulation, while some systems need elaborate checking of all the elements individually with the reactor shut down.

Most ruptured fuel detection systems use nuclear radiation monitoring systems which detect radiation which can be attributed to the release of fission products from the fuel element through the rupture in the clad. Some systems are based on the measurement of the increased pressure drop across the fuel channel caused by the physical deformation consequent on a rupture. The latter system is only an indirect monitoring system and has to be backed up by a radiation system.

Some of the main criteria that determine the sensitivity required of a ruptured fuel detection system are outlined below.

When a rupture is detected:

(*a*) The radiation levels near the reactor should be low for consideration of access and serviceability of reactor components;

(*b*) The contamination of reactor coolant and reactor components should be small for consideration of clean-up operations; and

(*c*) The rupture size should be small, so that it does not lead to a rapid rupture build-up before the reactor power can be brought down.

Some of the factors that limit the sensitivity of a ruptured fuel detection system are;

(*a*) Background signals in the monitor due to cosmic radiation, coolant activity, etc., and

(*b*) Uranium contamination on the surfaces of the fuel rods; this will show up as a rupture on the monitoring system.

Some of the radiation monitoring systems used for fuel rupture detection and their characteristics are given in Table 1. Work and results pertaining to an electrostatic precipitator using air, carbon-dioxide and water as fission-product carriers are reported here, giving the influence of various parameters in the design of such systems.

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Table 1. Typical ruptured fuel detection systems

Type of detection	Detector	Remarks
Gamma		
Gross gamma	G-M counter	High background due to parasitic activity in coolant (^{16}N, ^{41}A, ^{19}O, ^{14}C, etc.)
Gamma spectrometry of fission product nuclei of iodine, xenon, krypton, etc.	Scintillator	To improve discrimination against parasitic activity, any of the following methods may be used: *(a)* Collection of solid fission products on a particulate filter or on a graphite sponge; *(b)* In the case of liquid coolants, collection of fission products on anion ion-exchanger; *(c)* In the case of liquid coolants, the gaseous fission products may be scrubbed with a gas.
Beta activity due to gaseous fission products Xe and Kr.	Proportional counter, scintillator	Used widely in CO_2-cooled reactors; has high discrimination against parasite activity.
Alpha activity due to the release of uranium or plutonium.	Scintillator	
Thermal neutrons from delayed neutron precursors such as ^{87}Br, ^{137}I.	BF_3 counter, fission counter, boron-coated ion chamber	Background due to photoneutron sources, such as caused by ^{16}N activity and other neutron sources such as decay of ^{17}O.

THE ELECTROSTATIC PRECIPITATOR MONITOR

This system measures the activity due to two or more stages of beta emission; this activity is characteristic only of fission products and hence the discrimination against coolant activity is very good. This system has been extensively used in the carbon-dioxide cooled, natural uranium reactors [2] and the possibility of using it in other types of reactors [3] such as heavy-water reactors has been investigated to some extent.

The principle of operation of the precipitator monitor is briefly outlined here. The coolant sample is passed through a micron particulate filter and fed to a precipitator chamber where an electrostatic field collects all the ions formed by the decay of gaseous fission products in the coolant on the negative electrode. Some of the ions that have collected on the electrode decay further with the emission of a beta particle. The negative electrode is then brought under a shielded detector which counts the activity on the electrode.

Different geometries of precipitator chamber and mechanisms for moving the electrode beneath the scintillator head are in use [4]. Three of the common systems are shown in Figs. 1 a, 1 b and 1 c.

The precipitator monitor has to be contained in a pressure vessel which is compatible with the coolant gas and its operating pressure. It may also be necessary to cool the gas before admitting it into the precipitator monitor. Further, provision should be made to clean the accumulated activity on the scintillator detector and other components by means of a supply of fresh inactive coolant. One important consideration in the design of the precipitator chamber is to ensure that all the ions formed in the chamber are collected on the negative electrode [1].

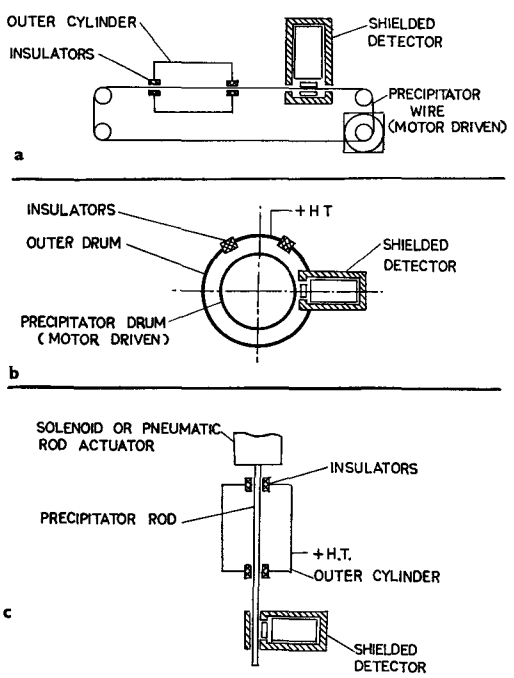


Figure 1.

(a): Wire precipitator; *(b)*: Drum precipitator; *(c)*: Rod precipitator

Considering a parallel-plate precipitator;

let n = ion density per cm^3

w_1 = volume flow rate of the gas per unit width cm^3/s cm

μ = ion mobility (cm/s)/(V/cm)

V = potential difference between the positive and negative electrodes (volts)

X = distance between the electrodes (cm)

Then, considering an element of width dx at a distance x from the negative electrode, the ion drift velocity $= v_d = \mu \frac{V}{X}$ (independent of x), ion migration time for distance x, $t_m(x) = x/v_d$, transit time $t_T = L/v_F = XL/w_1$ (independent of x) where L = length of the parallel plates.

All ions lying in a band $0 < x < X_c$ (where X_c is determined by the condition that ion migration time = transit time, i.e. $X_c/v_d = L/v_f$) = will be collected on the precipitator. For $X < X_c$ all ions flowing past the precipitator will be collected.

Ions collected, $q = nw_1$/s cm width,

$= nw_1 X_c/X$/s cm width.

Criticality condition: $X_c/v_d = L/v_f$ or $V/X_c = w_1/L$, i.e. $V = \frac{w_1 X_c}{\mu L}$.

Similarly the following critical condition can be obtained for the cylindrical precipitator:

$V = \frac{w \log_e R_c/R}{2\pi\mu L}$ assuming uniform axial velocity for gas flow, and

$V = \frac{w \log_e R_c/R}{4\pi\mu L}$ assuming parabolic axial velocity distribution for gas flow through the precipitator,

where W = volume flow rate of the gas through the precipitator (cm^3/s),

R = radius of the central negative electrode (cm) and

R_c = the radius of the precipitator chamber determined by the condition that ion migration time equals transit time.

COMPUTED PERFORMANCE OF AN ELECTROSTATIC PRECIPITATOR MONITOR

The parameters that are to some extent within the control of the designer of the detection system are:

t_d : the time delay between the fission-product entry into the coolant stream and its arrival at the precipitator chamber inlet,

t_T : the transit time of the coolant sample in the precipitator chamber,

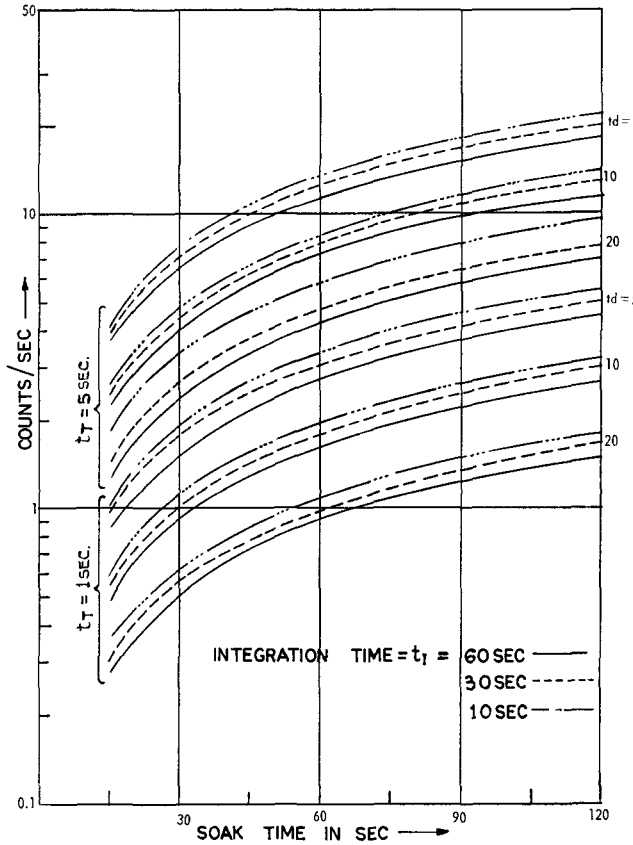


Figure 2. Wire precipitator performance. Computed results

t_s : the soaking time or the time for which precipitation is carried out (i.e. HT to the precipitator is applied),

t_x : the time delay between the completion of precipitation and the initiation of counting and

t_I : the counting time.

It is possible to predict the performance of the monitor [5] with variation in these parameters by considering the relevant fission-product decay schemes.

The activity counted on the detector is the cumulative yield of all the beta decays occurring in the fission-product chains starting with the krypton and xenon isotopes mentioned in the Appendix. The counting rates have been computed on the TIFRAC digital computer for a range of values of t_d, t_s, t_T and t_I and some of the results of these computations are shown in Fig. 2. All these computations refer to a fission rate of 100/s at the source and assumes 100% efficiency for precipitation and counting. Counting is also assumed immediately to follow the precipitation, i.e. $t_x = 0$.

The effect of the integration time on the count rate is not important and the time can be selected depending on the time for monitoring the fuel sample. The count rate decreases with increase in the time delay between generation of fission products and its reaching the monitor, and increases with increase in the transit time through the monitor. It can be said that the count rate

varies approximately as $1/t_d$ and t_T. The count rate increases with an increase in the soak time approaching saturation at a soak time of the order of a few minutes. This is illustrated in Fig. 5.

EXPERIMENTAL SET-UP AND RESULTS

Figure 3 shows the experimental set-up used for the measurements with the precipitator. A natural uranium disc 3.175 mm thick and 31.76 mm diameter placed in a beam hole in the Apsara reactor was used as a fission-product source.

The electrostatic precipitator monitor is an aluminium container 13.97 cm diameter and about 45.72 cm long. It houses the precipitator chamber, the scintillator photomultiplier assembly with the shield and the wire drive mechanism. The precipitator chamber has an outer cylinder 5.08 cm diameter and 7.62 cm long which is kept at a high voltage and a flexible wire 0.0794 cm diameter which is at ground potential. The length-to-diameter ratios smaller than this will give nonuniform electric fields whereas higher ratios will result in a precipitation of the fission-product activity over a length of wire much longer than that useful in counting. The wire is about 101.6 cm long and is formed into an endless loop which is driven over pulleys by means of a geared motor. Microswitches actuated by cams control the motor and move the wire by 19.05 cm, which corresponds to the distance between the centre lines of the precipitation chamber and the scintillator. The time taken for moving the wire over this distance is about 10 s. The time of precipitation or the soak time is controlled by a synchronous motor timer.

A controlled stream of air or carbon-dioxide is passed through the chamber containing the uranium disc and is then fed to the precipitator monitor through a particulate filter. The particulate filter is a micron filter and is assumed to trap solid fission products in clusters and prevent them from entering the electrostatic precipitator monitor. Solenoid-operated valves are used to admit either the gas containing the fission products or clean gas into the monitor. Admission of clean gas into the monitor can be used for purging the wire and scintillator of any activity collected on them. However it is advisable to have a separate purge inlet near the scintillator; this has not been provided in this assembly.

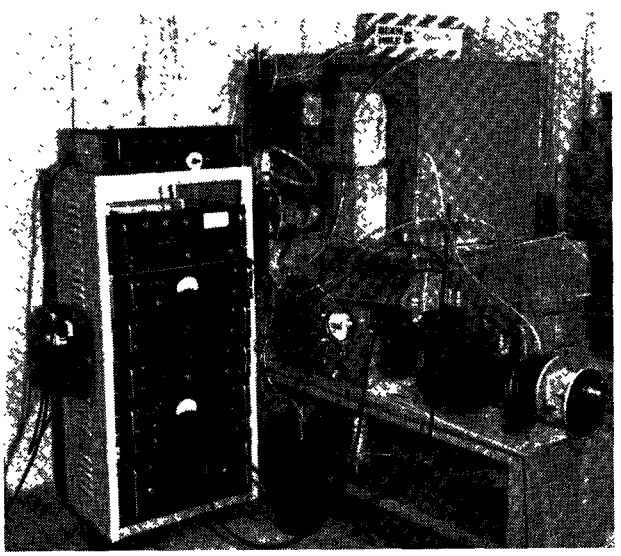


Figure 3. Experimental set-up

The detector uses a plastic scintillator 3.81 cm diameter by 0.635 cm thick, and the wire carrying the precipitated activity passes through a 0.3175 cm diameter hole in the middle of the scintillator. A thin scintillator has been used in order to reduce the counts due to gamma activity on the wire and in the vicinity of the scintillator. The preamplifier has a gain of about 60 and the single-channel analyser has been set for integral counting of all pulses above the discriminator level which had been kept at 3 volts.

Experiments were conducted using both air and CO_2 at atmospheric pressure and temperature, as fission-product carriers under the conditions mentioned below.

Gas flow: varying from 100 to 250 cm³/s, normally at 250 cm³/s,

Precipitation voltage: varying from 10 V to 4 kV and normally at 3 kV,

Precipitation chamber volume: 154.4 cm³,

Volume of piping between source chamber and precipitator: 435.4 cm³,

Air space volume inside source chamber: 31.6 cm³,

Thermal neutron flux at uranium disc: 1.5×10^{10} n/cm² s.

Saturation voltage for the precipitator. This precipitator is of the cylindrical type. Hence, the saturation voltage V (assuming parabolic axial velocity for gas flow) is given by $V = \dfrac{w \log_e R_c/R}{4\pi\mu L}$.

Under the experimental conditions,

$W = 250$ cm³/s, $R = 2.54$ cm, $R_c = 0.0397$ cm.

μ = ion mobility for fission products ions,

= 2.25 cm²/V s for air at atmospheric pressure and 300 °K,

= 1.48 cm²/V s for CO_2 at atmospheric pressure and 300 °K.

(The average value of μ for Kr and Xe ions in nitrogen at atmospheric pressure and 300 °K temperature is 2.33 cm²/V s and for other gases has been taken to be inversely proportional to the density of the gas).

Hence $V = 4.83$ volts for air and 7.34 volts for CO_2. Figure 4 shows the measured saturation curves for air and CO_2. It can be seen that the saturation voltages are several times higher than those calculated above. The discrepancy may be due to some extent to the

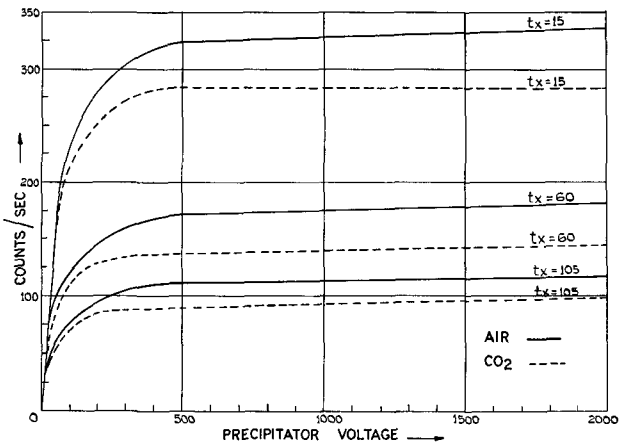


Figure 4. Wire precipitator monitor saturation characteristics

non-uniformity of electric fields inside the chamber, caused by gas inlet and outlet connections, the stranded central wire and finite length of chamber; it may be to a lesser extent due to errors in the value of ion mobility used. However it can be observed that the experimental curves demonstrate the phenomenon of saturation and also the difference in the saturation voltages between air and CO_2 because of the difference in ion mobilities in the two gases.

Precipitator monitor efficiency. An attempt has been made to obtain a figure for the over-all efficiency of the precipitator monitor assuming that the gas is able to scrub all the fission-product ions that come out of the uranium disc surface. At the temperature of the experiment (room temperature) the fission fragments that come out of the uranium disc are those due to recoil from the fissions in a surface layer of the sample of thickness equal to the fission fragment range; very few fission fragments will be able to diffuse from the inside of the uranium disc to the surface. It can be easily seen that the effective number of fission whose fission fragments are released outside the uranium disc is $\frac{1}{4}\Phi\Sigma_f A\delta$ where Φ is the thermal neutron flux at the uranium disc, Σ_f is the macroscopic fission cross section of the uranium sample, A is the exposed area of the uranium disc and δ is the fission-product range in uranium.

The measured value of Φ is 1.545×10^{10} n/cm^2 s; the number of effective fissions whose fragments will be released outside the uranium disc is 7.104×10^6/s.

Figure 5 gives the experimental results along with the calculated values. From these figures, it can be seen that the over-all efficiency (ratio of measured activity to calculated activity) of the electrostatic precipitator monitor in the current experimental set-up is about 2% and that assuming this value of efficiency the behaviour of the theoretical and experimental values with variations in t_s and t_I are in general agreement.

The count rate obtained in the above case for $t_s = t_I = 30$ s and $t_x = 15$ s per mm^2 of uranium area exposed per unit thermal neutron flux is 1.36×10^{-11} for air and 1.77×10^{-11} for CO_2. The difference in the count rates for the case of air and CO_2 is possibly due to the difference in their flushing abilities.

Background discrimination. Besides the count rate sensitivity per mm^2 area of uranium exposed per unit thermal neutron flux, the background signal will determine the minimum size of rupture that the system can detect. In the experimental set-up, it was seen that the reactor contributed significantly to the background, due to the proximity of the precipitator monitor to the reactor. In actual practice this background can be reduced by locating the precipitator monitor away from the reactor.

ELECTROSTATIC PRECIPITATOR MONITOR IN A WATER LOOP

As an immediate application for the electrostatic precipitator monitor developed during the course of this study, the feasibility of using it for fission-product detection in Apsara was investigated. For this purpose a water loop was set up with the same uranium disc type fission-product source used in the gas flow loop. The water containing the fission products is passed through a scrubber where it meets a flow of air. The air containing the fission products is passed through the electrostatic precipitator monitor. The results of this study are shown in Fig. 6.

The net efficiency of this system compared to the case of a primary gaseous carrier of fission products will be scaled down by the scrubbing efficiency of the fission products by the air stream. A good net efficiency can be obtained by suitable design of the scrubber, and optimising the water and the air flow rates for that scrubber. Experiments to study the performance of this system at very high air or water flow rates were not possible due to practical limitations. The ranges used in these experiments were for air 5 to 17 litres per minute and water 1 to 3 litres per minute. The scrubbing efficiency of the system was found to be about 50%.

It is proposed to install a similar set-up as a fission-product monitor for the swimming pool type reactor, Apsara, with facilities for individually scanning each fuel element if any offending signal is noticed by monitoring the bulk coolant flow as it comes out of the core top-face. This study is planned to be extended for other coolants such as heavy water, organics etc.

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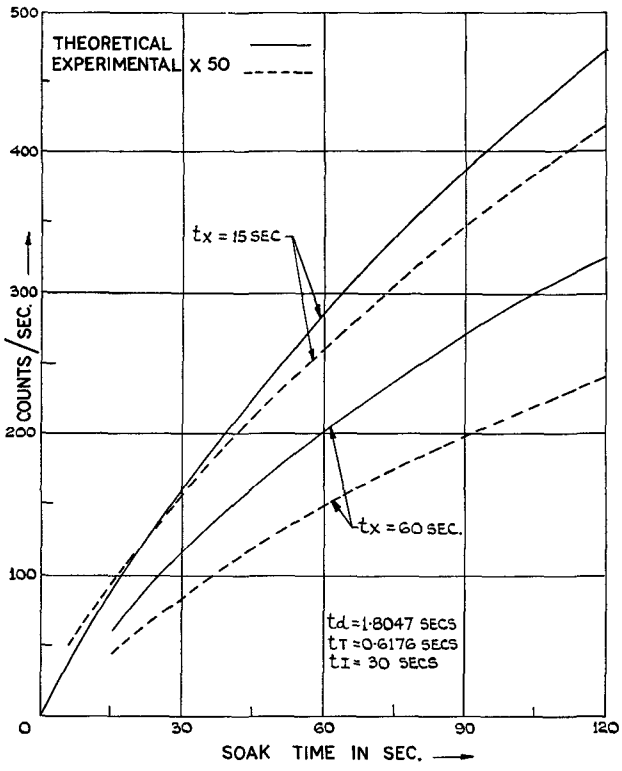


Figure 5. Wire precipitator performance

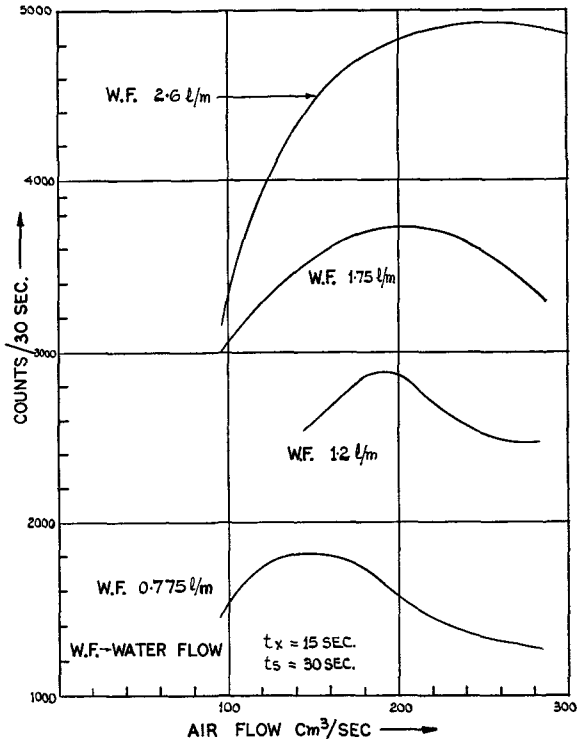


Figure 6. Water loop results (experimental)

to the staff of the TIFRAC computer, particularly to Miss B. G. Mythili, for assistance in programming and computing. The authors also wish to express their gratitude to Shri A. S. Rao, Director, Electronics Group, Atomic Energy Establishment, Trombay, for the support and encouragement extended in the work reported.

APPENDIX
Expressions for the wire activities

The following are the gaseous fission products that are relevant to the electrostatic precipitator monitor systems: ^{88}Kr, ^{89}Kr, ^{90}Kr, ^{91}Kr, ^{92}Kr, ^{93}Kr, ^{94}Kr, ^{138}Xe, ^{139}Xe, ^{140}Xe, ^{141}Xe and ^{143}Xe. The decay chains of these fission products together with their cumulative yields are available in standard references and the counts from each decay have been computed as shown below.

By examination of the fission product chains it is seen that they are of (or can be approximated to) three types:

(*a*) Kr → Rb → Sr (or Xe → Cs → Ba),

(*b*) Kr → Rb → Sr → Y (or Xe → Cs → Ba → La),

(*c*) Kr → Rb → Sr → Y → Zr
(or Xe → Cs → Ba → La → Ce).

It will be assumed that the particulate filter traps all the solid fission products and other particulate matter and allows only the gaseous fission products to pass through. As this filter is placed at the inlet of the precipitator monitor, precipitation happens only when krypton (or xenon) decays to rubidium (or caesium) inside the precipitator volume.

Counts from fission-product chains of type (a)*:*

Let Q = atoms of krypton released per second into the coolant gas,

P = the number of rubidium atoms per second that are deposited on the wire, assuming 100% precipitation efficiency,

$N_{Rb}(t_s)$ = the number of rubidium atoms left on the wire at the end of soak time t_s seconds,

$C_{Rb \to Sr}$ = the number of counts, counted over an integration time t_I after a delay time t_x between precipitation and counting, assuming that the entire precipitated portion is monitored.

From the above definitions, it can be seen that

$$P = Qe^{-\lambda_{Kr} t_d}\left[1 - e^{\lambda_{Kr} t_T}\right],$$

$$N_{Rb}(t_s) = \frac{P}{\lambda_{Rb}}\left[1 - e^{-\lambda_{Rb} t_s}\right] \quad (1)$$

and $$C_{Rb \to Sr} = \frac{P}{\lambda_{Rb}}\left[1 - e^{-\lambda_{Rb} t_s}\right]\left[e^{-\lambda_{Rb} t_x}\right]\left[1 - e^{-\lambda_{Rb} t_I}\right]. \quad (2)$$

Counts from fission-product chains of type (b):

In this type of decay chain, besides the counts due to those given for type (*a*) there will be counts due to:

(*i*) Decay of the strontium atoms to yttrium after the decay of rubidium atoms to strontium during integration time t_I i.e., $[C_{Sr\rightarrow Y}]$ from $N_{Rb}(t_s)$.

(*ii*) Decay of strontium atoms left on the wire at the end of $(t_s + t_x)$ seconds to yttrium while counting i.e. from $N_{Sr}(t_s + t_x)$ $[C_{Sr\rightarrow Y}]$.

Counts from (*i*): This can be shown to be given by

$$[C_{Sr\rightarrow Y}] \text{ from } N_{Rb}(t_s) = N_{Rb}(t_s)\, e^{-\lambda_{Rb} t_x}\left[\left(1-e^{-\lambda_{Rb} t_I}\right) - \frac{\lambda_{Rb}}{\lambda_{Sr}-\lambda_{Rb}}\left(e^{-\lambda_{Rb} t_I} - e^{-\lambda_{Sr} t_I}\right)\right]. \quad (3)$$

Counts from (*ii*): $N_{Sr}(t_s)$ = the number of strontium atoms left on the wire at the end of t_s seconds precipitation.
It can be seen that

$$N_{Sr}(t_s) = \frac{P}{\lambda_{sr}}\left[1-e^{-\lambda_{Sr} t_s}\right] + \frac{P}{\lambda_{Sr}-\lambda_{Rb}}\left[e^{-\lambda_{Sr} t_s} - e^{-\lambda_{Rb} t_s}\right].$$

Counts from these strontium atoms decaying into yttrium while counting will be

$$= N_{Sr}(t_s)\, e^{-\lambda_{Sr} t_x}\left[1-e^{-\lambda_{Sr} t_I}\right]. \quad (4)$$

Counts during integration time t_I from the decay of the strontium atoms generated from the decay of $N_{Rb}(t_s)$ during the delay time t_x between precipitation and counting will be $= N_{Rb}(t_s)\left[1-e^{-\lambda_{Rb} t_x}\right]\left[1-e^{-\lambda_{Sr} t_I}\right]$. (5)

Hence $[C_{Sr\rightarrow Y}]$ from $N_{Sr}(t_s + t_x)$ will be the sum of (4) and (5),
i.e. =

$$N_{Sr}(t_s)\, e^{-\lambda_{sr} t_x}\left[1-e^{-\lambda_{Sr} t_I}\right] + N_{Rb}(t_s)\left[1-e^{-\gamma_{Rb} t_x}\right]\left[1-e^{-\lambda_{Sr} t_I}\right]. \quad (6)$$

The total counts from fission-product chains of type (*b*) will be the sum of those given by Eqs. (2), (3) and (6).

Counts from fission-product chains of type (c):

In this type of decay chain, besides the counts due to those given for type (*b*) there will be counts due to:

(*i*) Decay of yttrium atoms to zirconium after the decay of rubidium atoms to strontium and these strontium atoms to yttrium during integration time t_I i.e., $[C_{Y\rightarrow Zr}]$ from $N_{Rb}(t_s)$;

(*ii*) Decay of yttrium atoms left on the wire at the end of $(t_s + t_x)$ seconds to zirconium while counting i.e., $[C_{Y\rightarrow Zr}]$ from $N_Y(t_s + t_x)$;

(*iii*) Decay of yttrium atoms to zirconium after the decay of strontium atoms to yttrium from the amount of $N_{Sr}(t_s)$ during counting time t_I.

Counts from (*i*): This will be negligible compared to the other contribution and need not be taken into account.

Counts from (*ii*): Let $N_Y(t_s)$ = the number of yttrium atoms left on the wire at the end of t_s seconds precipitation. It can be seen that

$$N_Y(t_s) = \frac{P}{\lambda_Y}\left[1-e^{-\lambda_Y t_s}\right] + \frac{P}{\lambda_Y-\lambda_{Sr}}\left[e^{-\lambda_Y t_s} - e^{-\lambda_{Sr} t_s}\right] + \frac{P\lambda_{Sr}\left[e^{-\lambda_{Sr} t_s} - e^{-\lambda_Y t_s}\right]}{(\lambda_Y-\lambda_{Sr})(\lambda_{Sr}-\lambda_{Rb})} + \frac{P\lambda_{Sr}\left[e^{-\lambda_Y t_s} - e^{-\lambda_{Rb} t_s}\right]}{(\lambda_Y-\lambda_{Rb})(\lambda_{Sr}-\lambda_{Rb})}.$$

Counts from these yttrium atoms decaying into zirconium while counting will be

$$= N_Y(t_s)\, e^{-\lambda_Y t_x}\left[1-e^{-\lambda_Y t_I}\right]. \quad (7)$$

Counts during integration time t_I from the decay of the yttrium atoms generated from the decay of $N_{Sr}(t_s)$ during t_x

$$= N_{Sr}(t_s)\left[1-e^{-\lambda_s t_x}\right]\left[1-e^{-\lambda_Y t_I}\right]. \quad (8)$$

Hence $[C_{Y\leftarrow Zr}]$ from $N_Y(t_s + t_x)$ will be the sum of (7) and (8),

$$\text{i.e. } = N_Y(t_s)\, e^{-\lambda_Y t_x}\left[1-e^{-\lambda_Y t_I}\right] + N_{Sr}(t_s)\left[1-e^{-\lambda_{Sr} t_x}\right]\left[1-e^{-\lambda_Y t_I}\right]. \quad (9)$$

Counts from (*iii*): The decay of yttrium atoms to zirconium as a continuation of the decay of strontium to yttrium from $N_{Sr}(t_s)$ during counting time t_I i.e. $[C_{Y\rightarrow Zr}]$ from $N_{Sr}(t_s)$ can be shown to be

$$= N_{Sr}(t_s)\, e^{-\lambda_{Sr} t_x}\left[\left(1-e^{-\lambda_Y t_x}\right) - \frac{\lambda_{Sr}}{\lambda_Y-\lambda_{Sr}}\left(e^{-\lambda_{Sr} t_I} - e^{-\lambda_Y t_I}\right)\right]. \quad (10)$$

Hence the total counts from fission-product chains of type (*c*) will be the sum of those given by Eqs. (2), (3), (6), (9) and (10). Again the contributions from (10) will be quite small and can be neglected.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/791 Inde

Conception d'un moniteur à séparateur électrostatique pour la détection de ruptures d'éléments de combustible

par B. S. Prabhakar *et al.*

Le comportement d'un moniteur à séparateur électrostatique pour la détection de ruptures d'éléments de combustible a été étudié en simulant les conditions de rupture dans le réacteur Apsara avec l'air et CO_2 comme gaz entraîneurs. L'optimisation des paramètres a été indiquée et les méthodes utilisées dans la conception sont discutées. Des calculs ont été faits pour déterminer l'efficacité du système en tenant compte des chaînes pertinentes de produits de fission. On montre l'accord entre la théorie et les résultats expérimentaux. On a modifié le modèle pour le système utilisant l'eau comme entraîneur en se servant d'air comme agent d'épuration. On est en train d'installer au réacteur Apsara un moniteur basé sur ce modèle.

A/791 Индия

Конструирование регистрирующей системы с электростатическим фильтром для детектирования разрыва тепловыделяющего элемента

Б. С. Пребхакар *et al.*

Характеристики регистратора с электростатическим фильтром для детектирования разрыва тепловыделяющего элемента были изучены путем имитации условий разрыва на реакторе APSARA с использованием воздуха и двуокиси углерода в качестве носителя. Рассматривается оптимизация параметров и обсуждаются методы конструирования. Проведены расчеты для определения эффективности системы с учетом соответствующих цепочек распада продуктов деления. Показано согласие между теоретическими и экспериментальными результатами. Конструкция усовершенствована для системы с водяным носителем с использованием воздуха в качестве скрубберного газа. Регистратор, основанный на этой конструкции, установлен на реакторе APSARA.

A/791 India

Diseño de un dispositivo de detección de elementos combustibles rotos basado en un precipitador electrostático

por B. S. Prabhakar *et al.*

Con ayuda de aire y CO_2 como gases portadores se ha estudiado, simulando las condiciones de rotura en el reactor Apsara, el funcionamiento de un dispositivo de vigilancia basado en un precipitador electrostático destinado a la detección de elementos combustibles rotos. Se indican los valores óptimos de los parámetros y se examinan los métodos de diseño. Se ha calculado la eficacia del sistema teniendo en cuenta las cadenas pertinentes de productos de fisión. Se muestra la concordancia de los resultados teóricos con los experimentales. El diseño se ha modificado con miras a adaptarlo a un sistema con agua como portador y aire como gas depurador. Se está instalando en el reactor de Apsara un dispositivo basado en este modelo.

Graphite oxidation and reactor safety

By J. R. Beattie,* J. B. Lewis ** and R. Lind ***

The oxidation of reactor grade graphite in air has been studied experimentally and theoretically in Britain as part of a research programme on the safety of graphite-moderated power reactors. Although this research is also relevant to higher temperature reactors, the present paper is concerned with the temperature range 400–600 °C only and with implications on the safety of the CO_2-cooled Magnox-uranium reactor. In assessing the safety of these reactors it is customary to postulate a rapid depressurisation accident. For designs using a steel pressure vessel and separate heat exchanger shells, it is assumed arbitrarily that a bottom or inlet duct fractures, and that some air enters the system before the damaged circuit can be isolated. Reactor temperatures in the short term are unaffected by graphite oxidation and the design conditions ensure that safe can temperatures are not exceeded during this period. But the heat of oxidation could be sufficient to affect the course of temperatures subsequently over a period of several hours, and knowledge of this is required to guide emergency planning and the choice of counter-measures. The factors which affect graphite oxidation particularly in a reactor environment have been intensively studied, and their effects on reactor temperatures in the longer term have been calculated. These studies and the conclusions drawn are described in this paper.

SOME FUNDAMENTAL ASPECTS OF GRAPHITE OXIDATION

The thermal reactivity of the graphite in a nuclear reactor is measured by means of a standard monitoring procedure in which small specimens are removed from the reactor and oxidised in air at temperatures of 400 or 450 °C. In order to be able to predict the rates of oxidation of the graphite under other conditions, for example in an air-CO_2 atmosphere at various temperatures, it is necessary to have an understanding of the factors governing the kinetics of the reaction. Experiments have therefore been carried out at AERE Harwell to investigate some of the more important chemical and physical factors which influence the rate of oxidation and the heat released. These are described below. They include studies of the ratio of CO_2 and CO produced in the reaction and of the variation of oxidation rate with partial pressure of oxygen the "order of reaction". The dependence of reaction rate on temperature was also studied since values of the activation energy ranging from 30 to 60 k cal/g atom had been reported, and fundamental reasons were sought for the variability of reactivity and activation energy. Since graphite is a porous material, in-pore gas transport, whether by diffusion or by bulk flow, is an important factor in limiting oxidation, and, following the earlier diffusion control theories evolved in the UKAEA by Dodson [1] and Robinson [2], an improved theory covering both effects has been evolved and tested experimentally.

The order of reaction was investigated by oxidising small specimens of reactor graphite at temperatures of 575, 595 and 655 °C in oxygen-nitrogen mixtures containing between 2.5 and 20% oxygen and under a total pressure of 1 atm [3]. The order of reaction was found to be very nearly 0.60; the use of this value in the transport control experiments described below gave excellent agreement between theoretical predictions and observed results for experiments carried out in the temperature range 450 to 650 °C. Continuous measurements were made of the rate of formation of both CO_2 and CO so that the separate orders of reaction for the production of these two gases were obtained. These were also very nearly 0.6. The ratio of CO_2/CO formed was found to be independent of the oxygen partial pressure over the range examined, but as expected the proportion of CO increased with temperature. Quite large variations existed between the ratios observed with different specimens; for example, to 655 °C one specimen gave a ratio of 1.4 whereas in a series of earlier experiments at this temperature most values were close to 4.0. Because of the sample-to-sample variations, which appear to be due to the presence of catalytic impurities, it is desirable to assume a high value for this ratio and consequently to use as the heat of combustion that for the formation of CO_2, namely 94 k cal/g atom.

A series of experiments was carried out in which 32 specimens of reactor grade graphite were oxidised

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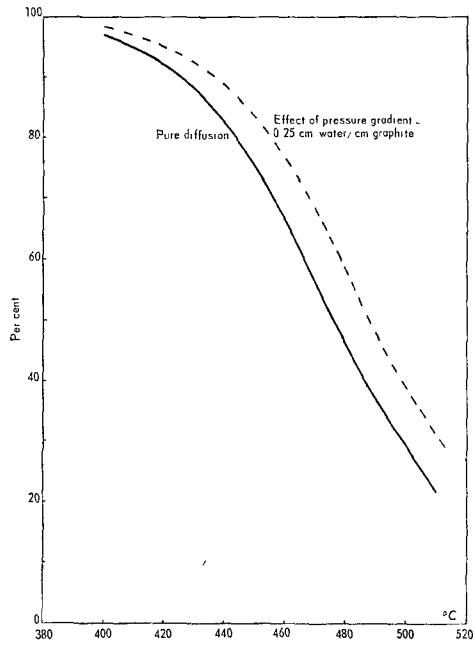


Figure 1. Utilisation efficiency (%) versus temperature (°C) for a tubular specimen of graphite

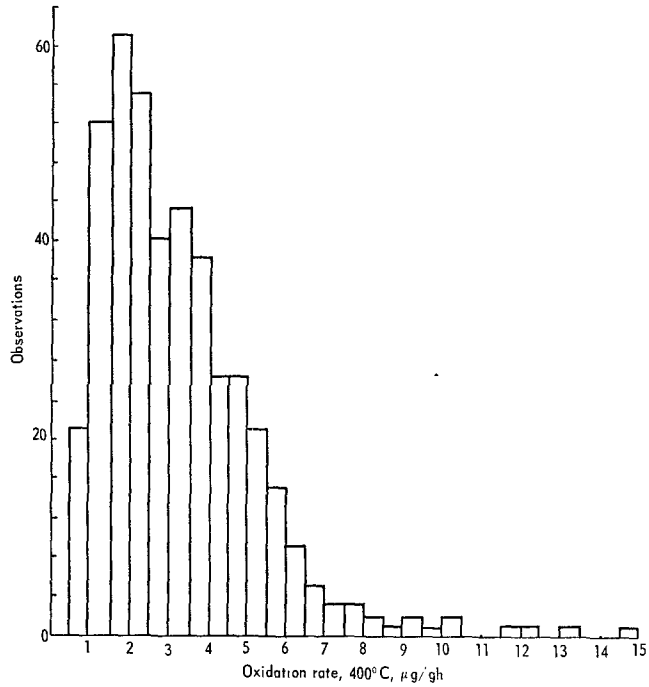


Figure 2. Histogram of air oxidation rates measured on outer slices of 429 blocks of "A" grade graphite

in normal air in the temperature range 450 to 550 °C, then ashed and the residues analysed spectroscopically. An attempt was made to correlate the observed reactivities and activation energies with the impurity contents. The reactivities varied by a total factor of 5 and the activation energies had a mean value of 43 with a standard error of 1. Although all specimens had a total ash content of about 200 ppm there was a wide variation in the level of specific impurities; for example, iron varied from 0.4 to 15 ppm and vanadium from 0.3 to 10. However, no correlation could be found between these values and either the rates or the activation energies. When spectroscopic grade graphite was oxidised in air containing less then 2 vpm moisture, a low oxidation rate and a high activation energy of 62 k cal/g atom were obtained. In normal air the rate was somewhat lower and the activation energy fell to 47 k cal/g atom. However when reactor grade graphite was oxidised in very dry air the results were very similar to those obtained in atmospheric air. It is clear that the low activation energies are connected with the impurity content of reactor graphite but no simple correlation exists at the level of impurity present in such material.

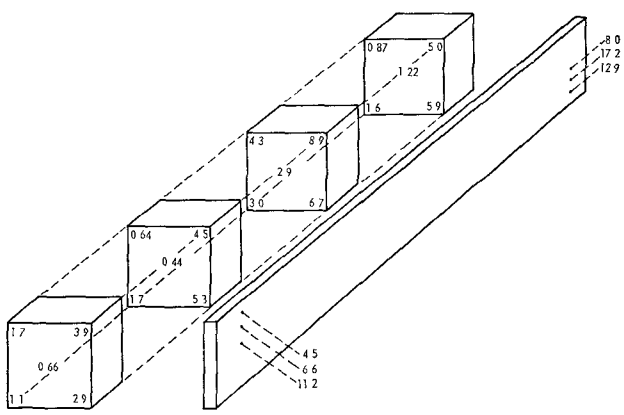


Figure 3. Variation of reactivity within a graphite block. Figures are oxidation rates in μg/gh measured at 400 °C

In the transport control experiments, tubular specimens of graphite 30 cm long, 3 cm id and of 6, 11.4, and 15.3 cm od exposed to air along the bore only, were oxidised at temperatures between 400 and 650 °C [3]. Under normal conditions transport to the internal pore structure of the graphite was by molecular diffusion from the bore, but by applying a pressure gradient across the tubular walls of the specimens bulk flow of air took place through the pores. The over-all rates of oxidation of the specimens were measured both under diffusional conditions and under conditions of superimposed bulk flow. It was found that there was transport limitation of the over-all rate of oxidation at all temperatures except the lowest, this limitation being markedly influenced by bulk flow. The results can be expressed graphically as in Fig. 1, where the surface utilisation factor η, which is defined as the ratio of the over-all rate to the chemical rate, and expressed as a percentage, is plotted against temperature for a tubular specimen of 15.3 cm od. Two curves are presented, that for pure diffusional transport and that obtained under an extremely small pressure gradient, ¼ cm water gauge/cm graphite. It can be seen that the very low flow rate associated with the latter pressure-drop changes η significantly. A mathematical model has

been developed with which the combined effects of diffusional and bulk flow on the over-all oxidation rate can be calculated for any practical condition. This model has been checked experimentally and shown to agree well with the observed data if the order of reaction is taken as 0.6.

VARIATIONS IN REACTIVITY OF GRAPHITE FROM PRODUCTION SAMPLES

The variations of graphite reactivity in production material and the changes of reactivity which are brought about by reactor operation have been studied in the UKAEA's Reactor and Production Groups, and these investigations together with some experimental work on inhibitors will be described.

Production material "A" grade graphite is monitored for air reactivity on a routine basis. A slice 1 in thick is machined from a block removed from a corner position in the graphitising furnace, and six specimens from the slice, each weighing about 10 g, are subjected to an air oxidation test at 400 °C. Results now extend to one block from each of 429 production heats, and are summarised as a histogram (Fig. 2) in which each recorded observation is the mean of the six samples tested from the individual slices. Most of the results lie close to the mean value of 3.22 μg/gh (at 400 °C), but the most important aspect of this data is probably the incidence of high reactivity material. Defining the latter arbitrarily as graphite with reactivity in excess of twice the mean value, it can be seen that about 5% of the results in Fig. 2 fall into this category.

The reliability of such data must depend on whether the slice tested is representative of the remainder of the block, and whether the block is itself representative of the heat. Data relevant to both points are being obtained, but the number of results is at present limited. Figure 3 however, shows the reactivities measured on specimens machined from a number of positions in a block for which the reactivity of the outer slice was found to be high. This reveals that the specimens from this slice were in general more reactive than those from the remainder of the block, although one side of the latter was rather more reactive than the rest. This result is typical of the small number of blocks so far examined for which the outer slice was reactive.

THE EFFECT OF REACTOR OPERATION ON REACTIVITY

An extensive programme of measurements and tests has now shown that exposure in an operating reactor causes a gradual change in the reactivity of graphite as subsequently measured in air. Information regarding this effect has been obtained from two sources, the Calder reactors themselves and irradiation tests mounted in high flux reactors.

The Calder reactors have provided information on graphite irradiated to a dose of ≈ 5 000 MWd/adjacent tonne maximum. The specimens which have been tested were either exposed in stringers located in selected channels, or trepanned from the channel walls. The scatter in the data has made the pattern of the results difficult to discern, although their aggregate clearly indicates an enhancement of reactivity. In the case of the trepanned samples for example, currently this amounts to a factor of about 5 relative to a typical virgin value. It is possible that several processes contribute to this enhancement. For example, the fast neutron component of the reactor radiation may create fresh active sites, radiolytic attack by CO_2 may increase the surface available for oxidation by air, and the coolant may in some manner "condition" the graphite (e.g. in respect of absorbed species) so as to increase its reactivity. It may be noted however, that contamination is unlikely to be a major contributor to the increased reactivity displayed by the Calder specimens, since graphite samples exposed in the out-of-pile section of the coolant circuit showed no significant increase in reactivity.

The second type of result which has been obtained is from air oxidation tests carried out on graphite samples which had been irradiated in hollow fuel element rigs as part of the Reactor Group graphite physics programme. These were irradiated in helium (not specially purified however) to Calder equivalent doses of ≈ 24 000 MWd/adj. t maximum. The results are subject to a number of uncertainties, the principal one probably being the effect of adventitious contamination on the outer surfaces of the specimens. Results for samples skimmed to remove, as far as possible, such contaminated material, suggested that the effect of irradiation at 250 °C was less marked than at 225 °C, and amounted at the former temperature to an increase of about 0.4 μg/gh at 400 °C per 1 000 MWd/adj. t Calder equivalent dose.

With the present uncertainty about the processes which may contribute to increases in reactivity, it is difficult to predict the value likely to be reached by the moderator after prolonged reactor operation. Fundamental work is required in this field and is in progress. Meanwhile the empirical approach is being continued with a view to establishing, for example, how the increases which arise from reactor exposure depend upon the reactivity of the graphite prior to irradiation. Another object of these studies is to determine what allowance must be made for the changes in pore structure which are brought about through radiolytic attack by the coolant. From the work carried out so far it is found that there is an increase in reactivity which reaches a maximum value when the graphite has lost a small percentage in weight, and the maximum effect is believed to be a four times increase in reactivity. At present the preferred method of estimating

the long-term reactivity is to use the data from the hollow fuel element rigs, modified by a factor which allows for the effect of radiolytic attack by the coolant. On this basis the general level for the reactivity of the moderator of a Magnox station at the end of the reactor life could be expected to be of the order of 100 μg/gh. at 400 °C.

INHIBITION OF GRAPHITE OXIDATION

Since the reactivity of the moderator after twenty or more years operation cannot as yet be predicted with certainty, the possibility cannot be excluded that the values ultimately reached (in localised regions, if not generally) might be hazardous. However, as is discussed later, it should prove possible to control a runaway condition by injection of carbon dioxide. Nevertheless, in order to provide an additional safeguard for the future, the UKAEA has sponsored work at the International Research and Development Company Limited (Newcastle) aimed at establishing the feasibility of controlling the temperature transient arising from ingress of air by injection of a suitable inhibitor. A number of compounds were tested for this purpose, including sulphur dioxide, tri-methyl phosphate, carbon tetrachloride, thionyl chloride, and chlorine. None however, was found to be as promising as phosphorous oxy-chloride ($POCl/_3$), which produced the lowest inhibited oxidation rate. Furthermore, the effects of this inhibitor were found to persist after injection had ceased. The results in this respect were in marked contrast to those obtained using chlorine, for which the oxidation rate not merely rose to the value prior to injection, but actually exceeded this.

Figure 4 shows a typical set of results for $POCl_3$. In this example, the graphite was oxidised at 548 °C in a once-through stream of oxygen, and the oxidation rate measured continuously by monitoring the effluent for CO and CO_2 by means of infra-red gas analysers. The introduction of $POCl_3$ (about 5% by volume) is seen to be marked by a sharp peak in the production of CO and CO_2, which is believed to be due to a displacement of gas from the surface of the graphite by the inhibitor. There is however, no significant production of heat over this period. Subsequently, it is seen that the rate falls to about one-seventh of the original value, and rises only slowly when addition of the inhibitor is discontinued. It is considered likely that the mechanism by which such inhibition occurs is the chemisorption of a phosphorous-containing species on the accessible surface of the graphite with consequent blocking off of active sites. The pick-up of phosphorous during these experiments has been found to be of the order of 100 ppm.

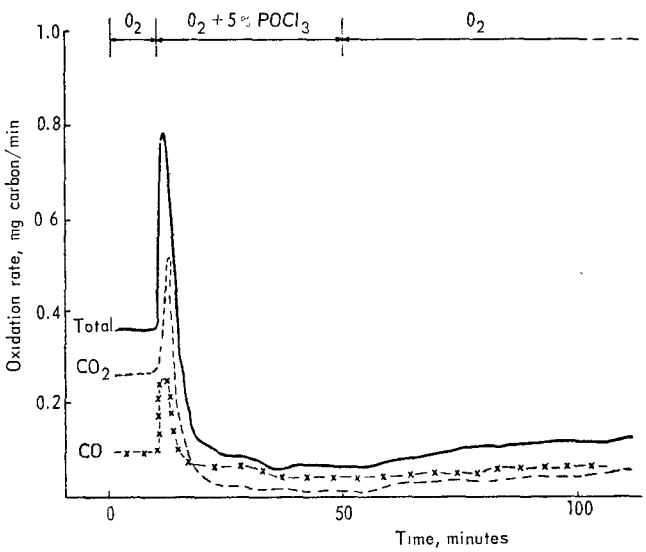


Figure 4. Effect of phosphorous oxychloride on oxidation of graphite at 548 °C

Further work is required to establish the effects of specimen size and to check the efficiency of the inhibitor for contaminated and irradiated graphite. It is also necessary to check that the compatibility of $POCl_3$ with critical components, such as fuel cans, is adequate. Preliminary work on some of these items suggests that $POCl_3$ will meet these further requirements.

THE INFLUENCE OF GRAPHITE OXIDATION ON REACTOR SAFETY

Some general conclusions regarding reactor safety may now be drawn from these investigations of graphite oxidation. The effect on temperatures following a hypothetical depressurisation accident is usually assessed using a digital computer to solve differential equation which describe the thermal behaviour of the reactor. The essential framework of such calculations consists of a set of equations each of which represents for an elemental length of reactor channel and a particular radial zone of fuel, coolant, or moderator, the balance between heat generated within the zone, heat stored (temperature rise) and heat transferred by radiation, convection and conduction. It is usually assumed that the main circulator motors trip and allow the circulators to run down to a lower speed which is then maintained by pony motors. Since some air may enter the system before the damaged circuit can be isolated, an assumption usually made in these studies, which embody a considerable degree of pessimism, is that the reactor coolant is instantaneously replaced by air after the duct fracture. During the period immediately following the fracture remedial actions would be taken, including isolation of the heat exchanger circuit containing the fractured duct, injection of CO_2 at a high rate, and restarting of main circulator motors in the undamaged circuits. An objective of exact calculation is to verify that these measures would be effective.

Following depressurisation, the initial rate of rise of graphite temperature is caused principally by heat which is thermally radiated from the fuel can. This originates as fission product decay heat, and a margin of safety is allowed by taking, for the total decay heat generated in fuel and moderator, figures which correspond approximately to 1.4 times the values quoted by Stehn and Clancy [4]. The fraction of decay heat which is absorbed directly as gamma rays by the moderator is small but significant. If the graphite temperature were permitted to rise too high, the heat of oxidation could strongly affect the latter part of the curves since the reaction rate increases rapidly with temperature. The heat of oxidation is represented by an Arrhenius equation with an activation energy of 43 k cal/g atom as previously mentioned. A range of possible values of reaction rate as measured at 400 °C can be considered since this will vary with reactor exposure. As an elementary calculation shows, the heat produced by oxidation of unirradiated graphite (mean reactivity about 3×10^{-6} g/gh at 400 °C) would be small compared to the radiated fission product after-heat, and would therefore have little effect on reactor temperatures. Greater interest is attached to the study of effects of higher reactivities, such as 1×10^{-4} g/gh a probable value for long irradiated material, since in such cases reactor temperatures might be markedly affected by the heat of oxidation. Some reduction in over-all reaction rate may be expected from the effect of in-pore diffusion control provided bulk flow of gas through the graphite is negligible. In earlier calculations, allowance has been made for the effect of in-pore diffusion by using a correction factor based on the depth of oxidation as defined in the theory of Robinson [2], but following the Harwell work referred to previously in this paper a more detailed calculation with some allowance for bulk flow could now be used. The calculations also take account of the variation of reaction rate with the 0.6 power of oxygen concentration in the coolant, so that one can allow for specified rates of emergency CO_2 supply to the reactor vessel, and, if required, for the effects of removal of oxygen by reaction with graphite. The latter effect, however, is generally small at the temperatures of interest here. The release of Wigner stored energy is allowed for, but this effect is also quite small.

To illustrate the behaviour of reactor graphite temperatures and the general principles which have been evolved, some temperature-time curves were calculated in a simple manner treating the graphite as a single radial zone. Although some useful general conclusions and ideas may be drawn from the curves obtained, they should not be taken as representing in more than a qualitative manner the behaviour of reactors. If exact calculations were required for a particular reactor, a more detailed treatment allowing for more than one radial zone of graphite would be required as well as the input data appropriate to that reactor. Figure 5 shows the effect of an extreme value of reactivity 1×10^{-4} g/gh as measured at 400 °C, corresponding approximately to material at the end of the nominal reactor life. In curve (*a*), pony motor flow is assumed and an allowance is made for the effect of diffusion control, and for purposes of illustration it is assumed that there is 100% air in the circuit throughout the time shown. For the case of pony motor flow, one may check that diffusion control would not be masked by bulk flow of gas by estimating pressure gradients across the reactor graphite and referring to the work on diffusion control described earlier in this paper. The curves show that with pony motor flow graphite temperatures would remain under control for at least several hours after duct fracture. Curve (*b*), which shows the effect of omitting diffusion control, is of theoretical interest only since, as stated above, pure diffusion control would operate for this low value of coolant flow. This illustrates the significance of the fundamental work aimed at establishing the reliability of this effect. In many cases pony motor flow would suffice to control the accident and ultimately cool the graphite and fuel. However at the higher levels of reactivity or with higher starting temperatures there is a greater risk of graphite temperatures continuing to rise towards the end of this period, and in order to

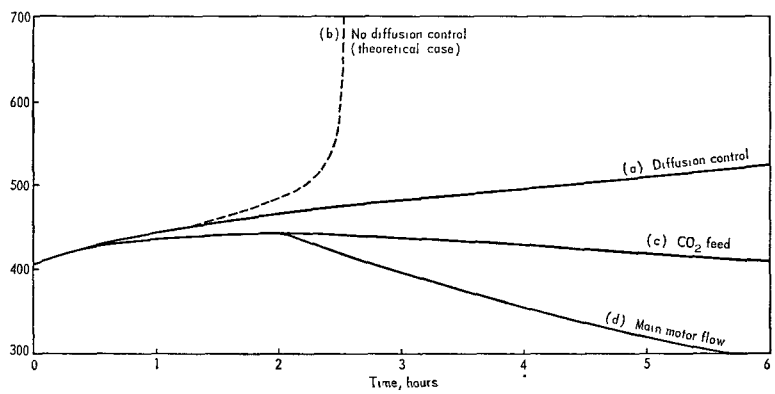


Figure 5. Maximum graphite temperature versus time

increase the margin of safety it is the practice to provide a large feed-rate of CO_2 for emergency use. This would be used as soon as possible after a depressurisation accident. But since it would take some time to displace the air from the vessel and circuit, the lengthy period of low graphite temperatures obtained under pony motor flow is of considerable value. Curve (*c*) shows the effect of CO_2 feed which leads utlimately to lower graphite temperatures. A more rapid reduction of graphite temperatures occurs if main motor circulator flow is restored and curve (*d*), showing the effect of this, is typical. Diffusion control was neglected for this case since estimates of possible pressure gradients showed that the effect might be partially masked by bulk flow of gas.

The possible effects of isolated regions of higher than usual reaction rate are being investigated theoretically, by making allowance for heat conducted axially through the graphite and heat radiated and conducted radially outwards to adjacent graphite blocks of normal reactivity and temperature. It is found that such a region, if it existed, would not give rise to high temperatures unless its length exceeded a critical value which depends on coolant flow, reactivity, position in the channel and other parameters. For example, some preliminary results suggest that regions of up to ten times higher reactivity and up to about 40 cm in length would be rapidly cooled by main circulator flow in air. In the low flow regime, 10 cm might be a critical length but further work on the influence of diffusion control and other aspects is required before this can be confirmed. This work, which at present is mainly of theoretical interest, should ultimately assist the evaluation of results from the graphite reactivity sampling programme as these are obtained.

CONCLUSION

In the work described the intention has been to gain an understanding of fundamental aspects of graphite oxidation, to obtain information on reactor graphite in the operational state, from this to predict the behaviour of reactors in an accident, and to use this knowledge to help devise any safeguards thought necessary. It is known that the reactivity of new reactor graphite presents no special hazard, but there is inevitably some uncertainty about conditions which might arise during a period of up to, say, twenty years from now. From the studies described it is believed that, over the range of reactor and graphite conditions likely to be encountered during the nominal life of the reactor, safe procedures can be evolved by making appropriate use of CO_2 feed and coolant flow conditions. This belief relies partly on extrapolation from experimental conditions which will not be finally confirmed until many more years experience of power reactor operations have been achieved. However, there is every prospect that the measures developed could be extended if required, and the encouraging results obtained from chemical inhibitors suggest that an additional safeguard would be available at these later stages if required.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/185 Royaume-Uni

Oxydation du graphite et sûreté des réacteurs

par J. R. Beattie *et al.*

Les études concernant l'oxydation thermique du graphite dans l'air ont formé une partie importante des recherches sur la sûreté des réacteurs au Royaume-Uni. La chaleur engendrée par l'oxydation du graphite pourrait affecter le résultat d'un accident, considéré comme possible pour les réacteurs Magnox modérés au graphite, qui serait dû à la rupture d'une conduite de fluide de refroidissement, ayant pour conséquence une rapide baisse de pression et la possibilité de l'entrée d'air.

Des expériences ont été faites pour étudier l'effet des impuretés et autres facteurs fondamentaux sur la rapidité de la réaction et ses variations avec la température. Dans la gamme des compositions de gaz ayant un intérêt pour cette étude, on a trouvé que le taux de réaction variait avec la puissance 0,6 de la concentration d'oxygène et que le produit de la réaction était en grande partie CO_2. On a étudié, dans d'autres expériences utilisant de gros blocs, l'effet limitatif de la diffusion gazeuse dans les pores du graphite et les résultats ont confirmé un modèle mathématique amélioré dans lequel il est tenu compte des effets combinés de la diffusion et du débit masse. La théorie montre que le débit masse produit par un gradient de pression d'environ 0,25 cm d'eau par cm de graphite de qualité nucléaire provoquerait des changements marqués dans les vitesses de réactions observées.

Des prélèvements de graphite provenant de lots de production ont montré qu'il existait une certaine variabilité dans le taux de réaction entre de petits spécimens. Cependant il semble vraisemblable que pour de gros blocs, la vitesse générale de réaction restera proche de la valeur moyenne d'environ 3×10^{-6} g/hh à 400 °C. Des échantillons extraits des modérateurs de réacteurs en fonctionnement et des spécimens exposés dans des réacteurs ont indiqué que le taux de réaction pouvait être multiplié par 10 au moins après exposition prolongée dans des conditions d'utilisation en réacteur et il est possible d'établir la relation entre l'augmentation, d'une part, et la dose d'irradiation et la perte de poids provoquée par l'exposition au fluide de refroidissement du réacteur, d'autre part. Dans d'autres travaux expérimentaux que l'on décrit, on a étudié le comportement de produits chimiques en tant qu'inhibiteurs de l'oxydation du graphite; l'un d'eux, l'oxychlorure de phosphore, abaisse le taux de réaction à une valeur au moins 10 fois plus petite si on l'injecte en petites quantités en cas d'accident.

On met en évidence l'influence de certains des facteurs ci-dessus sur le cours des événements suivant l'accident supposé en utilisant des courbes calculées température/temps pour un canal de réacteur et on explique les méthodes de calcul. L'effet de l'oxydation du graphite sur les températures de réacteur dépend fortement du taux de réaction supposé. La valeur pour une matière vierge, 3×10^{-6} g/gh, mesurée à 400 °C, a un effet négligeable, mais des valeurs beaucoup plus grandes que celle-ci pourraient conduire à des élévations de température du graphite. Les mesures de sécurité prévues pour un tel accident comprennent une amenée massive de CO_2 et on montre que ceci, combiné avec une circulation à la pression atmosphérique, pourrait empêcher des dégâts au combustible et la libération de produits de fission à la suite d'un tel accident. Les inhibiteurs chimiques pourraient constituer une utile sauvegarde complémentaire.

A/185 Соединенное Королевство

Окисление графита и безопасность реактора

Дж. Р. Бетти *et al.*

Изучение термического окисления графита в воздухе составляет важную часть Британской программы исследования безопасности реакторов. Тепло, генерируемое в процессе окисления графита, может привести к аварийной ситуации, характерной для магноксовых реакторов с графитовым замедлителем, например к аварии, связанной с появлением трещин в канале теплоносителя, что приводит к быстрому падению давления и возможности выброса теплоносителя в атмосферу.

Были проведены эксперименты по исследованию влияния примесей и других фундаментальных факторов на скорость реакции и ее изменение в зависимости от температуры. Для интересующих композиций газа было найдено, что скорость окисления пропорциональна концентрации кислорода в степени 0,6 и что основным продуктом реакции является углекислый газ. В других экспериментах с большими графитовыми блоками исследовались эффекты, ограничивающие диффузию газа из пор графита. Полученные результаты подтвердили выбор усовершенствованной математической модели, которая учитывает комбинацию диффузионных эффектов и переноса массы. Теория показывает, что перенос массы графита реакторной чистоты, обусловленный градиентом давления порядка 0,25 см вод. ст. на 1 *см*, должен приводить к заметным изменениям для наблюдаемых скоростей реакции.

Испытания графита, взятого из промышленных партий, показали, что у небольших образцов наблюдается некоторое различие в скорости реакции. Однако для больших блоков общая скорость реакции при температуре 400° С составляет $3 \cdot 10^{-6}$ *г/г · час*. Образцы, вырезанные из замедлителя действующих реакторов, и образцы, облученные в реакторах, показали, что при длительном облучении в реакторных условиях скорость реакции может возрасти по крайней мере на порядок и что может быть определено соотношение между ростом скорости, дозой облучения и весовыми потерями, обусловленными теплоносителем реактора. Описаны другие эксперименты, в которых изучались характеристики химических соединений, используемых в качестве ингибиторов окисления графита. Одно из таких соединений — хлорокись фосфора — уменьшает скорость реакции по крайней мере на порядок при введении ее в небольших количествах во время аварийных случаев.

Влияние некоторых из вышеуказанных факторов на возможность возникновения аварийных ситуаций иллюстрируется графиками зависимости расчетной температуры от времени для реакторного канала; описаны методы расчета. Влияние окисления графита на реакторные температуры в значительной степени зависит от выбранной скорости реакции. Скорость реакции для чистого материала $3 \cdot 10^{-6}$ *г/г·час*, измеренная при температуре 400° С, не играет значительной роли, однако гораздо большие величины могут привести к подъему температуры графита. Для ликвидации такого аварийного состояния потребуется подача большого количества углекислого газа. Показано, что подача газа в сочетании с потоком, циркулирующим при атмосферном давлении, предотвратит разрушение тепловыделяющих элементов и выделение

продуктов деления, сопровождающих такого рода аварию. Использование химических ингибиторов должно обеспечить дополнительную безопасность эксплуатации.

A/185 Reino Unido

La oxidación del grafito y la seguridad de los reactores

por J. R. Beattie *et al.*

Una parte importante de las investigaciones realizadas en Gran Bretaña sobre la seguridad de los reactores se refiere a estudios de la oxidación térmica del grafito en el aire. El calor generado por la oxidación del grafito podría alterar las consecuencias de un accidente como el que se ha postulado para los reactores Magnox moderados por grafito, en el que se supone que la ruptura de un conducto de refrigerante ocasiona una rápida disminución de presión y permite la entrada de aire.

Se han efectuado experimentos para investigar el efecto de las impurezas, de la temperatura y de otros factores fundamentales sobre la velocidad de la reacción y se ha encontrado que, en el intervalo de composiciones de gas que interesa, esta velocidad puede variar con la potencia 0,6 de la concentración del oxígeno y que el producto de la reacción es principalmente CO_2. En otros experimentos con grandes bloques de grafito se ha estudiado el efecto restrictivo de la difusión gaseosa en los poros del grafito y los resultados confirman un modelo matemático perfeccionado en el que se tienen en cuenta los efectos combinados de la difusión y del flujo másico. La teoría muestra que el flujo producido por un gradiente de presión de unos 0,25 cm de columna de agua por cm de grafito de calidad nuclear ocasionaría alteraciones notables de las velocidades de reacción que se han observado.

Entre muestras pequeñas de grafito tomadas de diversas partidas de producción se ha observado cierta variabilidad en la velocidad de reacción. No obstante, par ce probable que para masas grandes la velocidad media de reacción sea próxima al valor de 3×10^{-6} g/gh a 400 °C. Muestras sacadas del moderador de varios reactores y otras intencionadamente expuestas han indicado que la velocidad de reacción puede aumentar por lo menos en un orden de magnitud después de una exposición prolongada, y es posible relacionar el incremento con la dosis de irradiación y con la pérdida de peso a que da lugar el contacto con el refrigerante del reactor. En otros trabajos experimentales descritos en la memoria se ha investigado la eficacia de productos químicos como inhibidores de la oxidación del grafito; uno de ellos, el oxicloruro de fósforo, inyectado en pequeñas cantidades en un momento de emergencia parece que podría reducir la velocidad de reacción por lo menos en un orden de magnitud.

La memoria ilustra la influencia de algunos de los factores indicados sobre los resultados del accidente postulado con la ayuda de curvas calculadas de temperatura/tiempo para un canal del reactor, y describe someramente los métodos de cálculo. El efecto de la oxidación del grafito sobre la temperatura del reactor depende en gran medida de la velocidad de reacción que se suponga. En cuanto al material virgen, el valor 3×10^{-6} g/gh medido a 400 °C, tiene un efecto despreciable aunque valores muy superiores podrían ocasionar una elevación de la temperatura del grafito. Una de las medidas para contrarrestrar este efecto consiste en la inyección masiva de CO_2 que, si además se hace circular a la presión atmosférica, evitaría los daños al combustible y la liberación de productos de fisión que un accidente de este tipo ocasionaría. Los inhibidores químicos constituirían una salvaguardia adicional de gran valor.

Aerospace safety of isotopic and reactor power sources

By G. A. Fowler *

The Systems for Nuclear Auxiliary Power (SNAP) which are being developed by the United States Atomic Energy Commission (USAEC) for various applications show considerable promise for providing electrical power for space missions. Radioisotope units in the power range from watts to hundreds of watts can provide compact, long-lived sources of electrical power unaffected by eclipses or the radiation environment of space. Their usefulness already has been demonstrated in pay-loads placed in orbit in 1961 and 1963 [1, 2]. The USAEC is proceeding with plans for improved radioisotope processing facilities that should markedly reduce the cost of the isotopes and ensure a supply adequate for all foreseeable needs. Where higher power is demanded, the reactor-type SNAP systems offer a compact source of electrical power in the range from hundreds of watts to megawatts. As yet, no reactor supply has been flown, but development is proceeding toward systems which can be flown. The USAEC has stated its interest in seeing the developments carried through the demonstration phase, including both ground and flight testing.

The SNAP systems derive their power from the heat generated by the spontaneous nuclear fissions of radioisotopes, or from the controlled nuclear fissions of a reactor. In either case, the basic source of the energy is from within the nuclei of the atoms of the material and hence is independent of chemicals, sunlight, magnetic fields, etc. SNAP systems can provide power for relatively long periods with high reliability and in this characteristic are ideal for space missions. The principal disadvantage of the SNAP systems comes from the same source as their principal advantage, namely that the system derives its power from radioactive materials that are potentially dangerous to human life. However, to deny the use of the SNAP systems for this reason alone would be to ignore the fact that every other source of power that has been developed to man's benefit also is potentially dangerous to its user. We have grown accustomed to the dangers present in our use of fire, steam, electricity and gasoline in our everyday life. There can be little doubt that we will ultimately accept radioactive power sources as commonplace. The speed with which they reach this status depends to a great extent on how smart we are in using them safely, particularly during the introductory phase.

Recognizing this, the USAEC has placed considerable emphasis on the safety aspects of the SNAP systems. First of all, the designer of the SNAP system is required, as part of the design responsibility, to provide a thorough safety plan for the SNAP system throughout its factory-to-orbit sequence. The responsibility for providing a safe system is placed squarely on the shoulders of the power supply designer. In addition, the Assistant Director for Nuclear Safety of the Division of Reactor Development makes an independent assessment of the adequacy of the safety plan for each SNAP system. This assessment involves a complete study of the safety plans for the SNAP system, as well as its proposed application to a space vehicle. It requires thorough analytical studies, and may involve both ground and flight testing. Sandia Corporation is the principal contractor for the USAEC for the aerospace nuclear safety program.

This paper will discuss some of the problems related to nuclear safety associated with launch pad accidents, launch aborts, and re-entry for space power plants of isotopic and reactor types. In addition, plans for research programs and the results of the flight test program are included. The paper is written on the basis of space technology as it exists today, but with the full appreciation on the part of the author that the next five years will bring about changes in the technology of space that may cause some of the present-day problems to disappear.

FACTORYS TO RE-ENTRY SAFETY

The safety plan for aerospace nuclear power supplies must cover all phases of the handling of the power supply from initial assembly through its launch and possible return to earth. The safety questions that arise during assembly and transportation to the launch area are generally within the scope of the industrial safety practices that the USAEC has developed over the past years. The testing that is required to simulate an accident throughout this portion of the nuclear power supply's career can be accomplished by more or less readily available facilities. The nuclear power supply (whether isotopic or reactor) is in a shipping

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configuration and is well protected against all foreseeable accidents. Shock, fire and immersion are the principal conditions which the shipping configuration must be able to meet without failure.

However, when the flight pay-load, including the nuclear power supply, has been mated to the launch vehicle and the launch count-down has begun, conditions are somewhat changed. The nuclear power supply no longer has the protection of its shipping configuration and is sitting on the top of a tall missile-shaped pile of explosive. At this stage, five major possibilities attract the attention of the safety analyst: (*a*) the launch vehicle may blow up and burn on the pad, (*b*) the vehicle may lift off successfully but blow up or be destroyed while still in the general vicinity of the launch area, (*c*) it may fail to boost the pay-load to orbital velocity, (*d*) it may achieve a short-lived orbit, with the pay-load returning to the earth in days or months, or (*e*) it may achieve a good orbit with the pay-load in orbit for years. In cases (*a*), (*b*), and (*c*) the isotopic supply will be at its highest activity level; in case (*d*) it will be partially decayed, and in case (*e*) it will be down to a relatively low level. The reactor supply will be at a low level for all cases except (*e*), assuming the reactor will be started only after it has been placed in a good orbit.

In the case of the launch pad explosion and fire, the items of particular interest are the overpressure seen by the nuclear power supply as a result of the explosion, the effect of the subsequent impact in the vicinity of the pad as the power supply falls or is thrown from the nose of the launch vehicle, and the effect on the power supply of the temperature rise due to heating from the fuel fire. If the system is an isotopic supply, the safety question is whether or not the fuel capsules will remain integral; if it is reactor supply, the safety question is whether or not there will be a nuclear excursion upon impact. Only incomplete information exists about the overpressure from the various launch pad explosions and about the temperature-time history of launch pad fires. Recognizing the need for this information, the Department of Defense (DOD), the National Aeronautics and Space Agency (NASA) and the USAEC have set up a co-ordinated launch-pad instrumentation program. Various types of active and passive measuring devices have been installed at selected launch sites, accommodating both liquid- and solid-propellant boosters, so as to be able to fill in the missing information should an accident occur [3]. The information obtained will allow environmental testing conditions to be set realistically without the penalty of severe overtesting or the false security of undertesting. Boosters are not blowing up on the pad the way they did in the past, so information is coming in very slowly from this program.

While the attention of the USAEC is focused primarily on the SNAP system design, the Commission also must concern itself with the manner in which the SNAP system physically is incorporated into the space vehicle, since this is of considerable importance to the over-all safety plan. The re-entry heating situation for a nuclear power supply that is buried inside a large flight vehicle obviously is different from that for a similar power supply that is mounted on the end of a boom and consequently is relatively independent of the flight vehicle during re-entry. Not only must a nuclear power supply be considered as an entity, but also it must be considered as a part of the system in which it is incorporated for flight so that the safety questions can be given adequate consideration.

If the missile leaves the launch pad successfully but fails or is destroyed early in flight, then the main sources of damage to the nuclear power supply are overpressure from the explosion and the consequences of the impact into water, soil, or surface structures. As in the case of the launch pad accident, the isotopic supply must be designed to remain integral, whereas the reactor supply must be designed so as not to go critical under these circumstances. Fire is less of a concern here, but impact velocities will have increased appreciably as compared to the launch pad accident, with the consequence of more severe impacts. One additional factor is that the impact area in this case will not be as well controlled from public access as is the launch pad.

The vehicle which falls just short of orbital velocity encounters essentially the same re-entry conditions as those encountered by an orbital decay; however, a significant difference is that the possible impact areas can be fairly well determined in advance. These potential impact areas can be kept over relatively unpopulated sea and land masses to minimize hazard under this failure condition. Also, an active control system aboard the flight vehicle is available for the command of actions relating to the nuclear power supply, if desired. Disassembly can be caused by mechanical or explosive means, the power supply can be cut loose from the flight vehicle, or other positive actions can be taken to improve the safety plan.

The short-lived orbit (days to months) still offers the possibility of active control of safety actions, but leaves the re-entry and impact locations up to chance. The long orbit (years) denies reliable active control (with present-day technology), has random re-entry and impact location, but offers time enough for the radioactivity to have decayed to a low level before re-entry occurs.

RE-ENTRY BURN-UP

For the last three of the five cases discussed above (suborbital, short-lived, and long-lived orbit), the flight vehicle and the nuclear power supply will be subjected to severe re-entry heating conditions. The

designer of the nuclear power supply is faced with the choice of designing for survival of the re-entry heating to allow for intact re-entry, or for the use of the re-entry heat pulse to cause burn-up and dispersal in the upper atmosphere.

Intact re-entry, particularly if it were controlled to impact into a designated area, is attractive from the viewpoint of nuclear safety for it essentially eliminates the major questions about public safety in the use of nuclear power supplies in space. Were this capability easy to achieve, the power supply could be returned to one of the existing land ranges and accurately tracked to impact. A recovery party could then go to the area and take whatever measures were indicated for the recovery and disposal of whatever radioactive material remained. Unfortunately, the controlled re-entry costs weight, and it is weight which must be taken from the pay-load and is chargeable to the nuclear power supply. In addition, the controlled intact re-entry calls for a reliable active command and retro-rocket system aboard the space vehicle. Today's technology does not offer much assurance that this would be available in the long-lived orbit case mentioned earlier; however, it could be accomplished with high confidence for the short-lived orbit. As we gain more experience with reliable electronics and with propellant life in space, and as other improvements in technology are made, reliability limitations will probably disappear, leaving only the extra weight as the price for controlled intact re-entry. Another disposal possibility which would exist, if reliability and weight presented no problem, would be to move the space vehicle containing the nuclear power supply to a long-lifetime orbit where it would remain for hundreds of years.

In each nuclear power supply being considered for orbital flight, the designer has chosen to use re-entry burn-up and dispersal in the upper atmosphere as the method for disposing of the radioactive material. This choice has caused the aerospace nuclear safety program to undertake the investigation of high-temperature materials properties, ablation droplet formation, and upper-atmosphere circulation patterns.

The problem of survival of re-entry heating, both on the relatively steep re-entry trajectories of the ballistic missile and the shallower re-entry of the manned orbital flights, has received expert attention over the past several years. On the other hand, the deliberate use of the mechanisms of orbital re-entry heating for the burn-up and dispersal of various materials in assorted configurations has been of limited interest, except to the designers of aerospace nuclear power supplies. As a result of the missile effort, it is possible with high confidence to calculate the heating rate on simple shapes for a given set of re-entry conditions, using any one of the several methods which have been developed from the work of NASA and various DOD contractors [4]. However, the irregular shape of the nuclear power supply, by comparison with nose cones, as well as the variety of different materials incorporated in the typical nuclear power source, present some interesting problems to the designer or the safety analyst when it comes to computing the effects of this heat in-put on the power supply. To determine the expected heat in-put on various parts of a complex geometry, it is necessary to use the hypervelocity or hypersonic shock tunnels such as those operated by the Arnold Engineering Development Center of the Air Force, or the Cornell Aeronautical Laboratories. In these tunnels, it is possible by full or fractional scale-model tests to establish the heat in-put relationships between the simple shapes for which computational models exist, and various parts of the more complex geometry of the specific nuclear power supply being examined. With these in-puts applied to a thermal model of the nuclear power supply it is possible, provided the material properties are known, to calculate the time of melting and disassembly of the various parts of the power supply when it is exposed directly to the re-entry heating pulse.

While shock tunnels can be used to provide short runs at high heat-transfer rates, there exists at present no facility that realistically can reproduce re-entry flight conditions for periods of time comparable to actual orbital re-entry. Such a facility would be of great benefit to the aerospace nuclear safety program. It would allow the rapid determination of the effect that a design change in the nuclear power supply would have on its burn-up characteristics. It would allow experiments such as the measurement of ablation droplet size distribution. It would be a powerful tool for the study of any re-entry heating phenomena. Unfortunately, there is no prospect that such a facility will be available soon. Judging by experience with the development of other environmental facilities, it is likely that any re-entry heating facility that it is practical to build and operate will do only a partial job of simulating the actual re-entry conditions, so that dependence upon analytical methods as well as actual flight testing will continue to be heavy. Nevertheless, a good re-entry heating facility capable of running for many minutes would materially assist both the nuclear power supply designer and the aerospace nuclear safety program.

RESEARCH PROGRAM PLANS

To assess the efficiency of the re-entry burn-up and dispersal characteristics of a nuclear power supply in a particular application, it is necessary first to examine the physical arrangement for its incorporation into the flight vehicle. The aerodynamic characteristics of the flight vehicle during re-entry into the earth's atmosphere must be determined so that the vehicle attitude

will be known and hence the nature of the re-entry heating to which the power supply will be subjected. From the nuclear safety viewpoint, a good vehicle design will place a SNAP device that is designed for burn-up in a position to be exposed to the full effect of the aerodynamic heating throughout its re-entry into the earth's atmosphere.

Once the character of the re-entry heating that will be applied to the nuclear power supply has been established, the power supply is "flown" in the computers by the aerospace nuclear safety organization to examine whether or not the heat available from the re-entry is sufficient to melt the container and allow the ablation of the radioactive material. At this point the need for better materials properties information has become apparent for many of the materials that are incorporated into nuclear power supply designs. The lack of complete materials properties data at and above the melting point has caused the aerospace nuclear safety organization to ask the Oak Ridge National Laboratory to compile the information available on various materials and to evaluate the capability throughout the country for undertaking the effort required to supply the needed information. The Division of Isotope Development of the USAEC also is actively supporting this effort. There are many properties of interest to this investigation, but the thermal and chemical properties are receiving first attention. Priority has been assigned to those materials that will be applied in SNAP systems most likely to be flown next. When the gaps in the required information have been identified, action will be taken to conduct the materials properties studies at the laboratories best equipped for the task. In the meantime, specific information needed for current studies is being obtained from investigation underway at Sandia Corporation and at the Battelle Memorial Institute.

If the re-entry heating calculations fail to show the complete melting of the radioactive material in the nuclear power supply during the re-entry heating period, then the burn-up and dispersal will not be accomplished. However, when melting of the fuel is calculated to occur, then the question of the size of the particles that will result and where they will be distributed in the upper atmosphere becomes of interest. At present, this is probably the area of greatest uncertainty in connection with re-entry burn-up studies. Steps are being taken to develop methods of determining particle size distributions under various conditions. The Cornell Aeronautical Laboratories [5] have undertaken studies for the aerospace nuclear safety program, and additional work is being done at Sandia Corporation. Experimental techniques developed at both places offer promise to provide some of the needed particle size information for materials of interest. Zirconium metal has been used in both laboratories during the development of the apparatus and the techniques. This material has exhibited the cooperative property of exploding promptly into many very small drops when molten droplets of a few hundred microns in diameter are exposed to the atmosphere. At this time it is not known how many of the other materials will exhibit similar properties, nor is it known what is the minimum size of droplet that will exhibit this explosive property. It is anticipated that information will develop rapidly in this area. In addition to the effort at Cornell and Sandia, the Illinois Institute of Technology has undertaken an investigation for the purpose of understanding and evaluating the possible coalescence of small-particle ablation debris to form larger masses of material [6].

When the particle size and distribution in the upper atmosphere by re-entry burn-up are known, the question remains as to when and where the material will reach the biosphere. While the general behavior is known for various ranges of particle sizes and altitudes, additional studies have been undertaken by contract to the Geophysics Corporation of America (GCA) and to Travelers Research Corporation (TRC). The GCA effort is to derive an analytical model to trace the fall and dispersion of particles injected at 250 000 ft altitude and falling to 100 000 ft. The TRC effort relates to the dispersion in the region of 100 000 ft altitude and below. Both of these efforts are primarily theoretical in nature but could lead to the development of an experimental program, if existing data prove inadequate to meet the needs of the aerospace nuclear safety program.

The research investigations outlined above will bring a better understanding of the materials properties, ablation droplet formation size, and distribution and upper-atmosphere dispersion of small particles. With this information, the analysis of re-entry burn-up for various proposed aerospace nuclear power supplies can be greatly improved.

RE-ENTRY FLIGHT TESTING

Flight testing is an important phase of the nuclear safety program, since it offers the opportunity for a much more realistic set of re-entry conditions than can be achieved in the laboratory. However, it is expensive and can provide only a relatively small amount of data per flight compared with the repetitive capability of laboratory testing. Therefore, the flight test is included in the program as the final demonstration of the safety features of a given nuclear power supply design. After analytical studies, computations and laboratory testing have been utilized to the fullest by the aerospace nuclear safety analyst in the assessment of the safety features, then flight testing is used as the final step in the process to demonstrate the safety characteristics of the nuclear power supply under re-entry conditions.

The first re-entry flight test (RFD-1) in the aerospace nuclear safety program took place in May 1963 [7]. The flight test was made in connection with the assessment of certain safety features of a proposed aerospace reactor power supply. The purpose of the re-entry flight was to verify the operation of design features that were intended to enhance the disassembly of the reactor under re-entry heating conditions, to verify temperature rise calculations at various points in the reactor mock-up during re-entry and, as a secondary objective, to obtain some gross indications of the rate of burn-up of full-scale models of the reactor fuel rods that were independently exposed to the re-entry heat pulse by ejecting them from the re-entry vehicle.

The re-entry carried a full-scale instrumented mock-up of the reactor assembly provided by the reactor designer and was equipped with a C-band radar beacon, an FM-FM radio telemetry system, and a 100-second delay tape recorder designed to permit data transmission following emergence of the telemetry signal from the ionization black-out caused by the re-entry heating. Data were transmitted continuously both in real time and with the 100-second delay. The reactor mock-up was instrumented with thermocouples for temperature-rise measurement and with action switches which would show when physical parts were removed from the mock-up as a result of re-entry heating. The re-entry vehicle contained a parachute and recovery aids, so that if it survived the re-entry it might be found and examined. Physical examination of the re-entry vehicle was desired to provide information of use to future re-entry vehicle designs, as well as to allow examination of the remains (if any) of the reactor mock-up. The re-entry vehicle was placed into the desired re-entry conditions (minus six degrees, 20 000 ft/s, at an altitude of 400 000 ft) by an NASA Scout vehicle. This four-stage solid-propellant system, which was launched from Wallops Island, Virginia, did a near-perfect job of placing the re-entry vehicle on the flat trajectory required to bring it into the range of the instrumentation located on the island of Bermuda as well as in various down-range aircraft.

Radio telemetry from the transmitter aboard the re-entry vehicle was received and recorded at NASA Wallops Island, at the NASA Mercury station on Bermuda, and aboard aircraft orbiting in the down-range area. Aircraft operated by the Air Force Missile Test Center, the Air Force Special Weapons Center, and NASA recorded the telemetry signal. Good radio telemetry signals were received from lift-off until ionization black-out of the telemetering signal occurred 350 seconds later at an altitude of 200 000 ft. No signal appeared after the black-out effect was over. It is believed that severe re-entry vehicle damage occurred during the black-out period when the peak heating rate was encountered. Radar tracking of the C-band beacon mounted in the re-entry vehicle was accomplished by NASA instrumentation radars at the Wallops Island station and at Bermuda. Excellent tracking was obtained, and the trajectory information was good until the ionization black-out blocked the radar beacon transmission at 170 000 ft, some 360 seconds after lift-off.

Optical observations were made by Sandia Corporation equipment installed on Bermuda Island. Tracking signals from the NASA FPS-16 radar on Bermuda were used to point two long-focal-length tracking telescopes in the right direction until the operator was able to acquire the re-entry and to track manually. In addition to the tracking cameras, fixed-plate cameras were mounted to cover the visible portion of the trajectory. Spectrographic and photometric coverage also were provided. Several of the aircraft carried cameras for photographing the re-entry. Unfortunately, the light level received at the Bermuda station was below even the most conservative estimates, so that the tracking telescope and the photometer film were useful only during the brightest part of the re-entry. Good records were obtained from the fixed-plate cameras and from the spectrographic instruments on Bermuda and aboard some of the aircraft. The NASA aircraft that was in a favorable position obtained a very useful movie film that covered the final part of the trajectory, as well as good spectrographic records.

The flight demonstrated that the disassembly features provided in the reactor design worked properly. The flight also verified that the temperature rises caused by re-entry heating of the reactor model were predictable with reasonable accuracy, and gave a gross indication of the rate of burn-up of the reactor fuel rods that were released from the re-entry vehicle shortly after fourth-stage ejection. Testing and additional analysis done on the re-entry vehicle design subsequent to the flight test indicate that there were areas of more severe localized heating on the re-entry vehicle than originally anticipated, and it is believed that the ablative covering was inadequate in some areas, so that burn-through occurred. It appears that the irregular shape of the reactor mock-up carried on the front end of the re-entry vehicle was a major contributor to the vortices that caused the excessive local heating.

A second re-entry demonstration flight, RFD-2, is planned for the fall of 1964. This flight will carry a model of an isotopic power supply which will be instrumented to investigate experimentally the disassembly features incorporated into the power supply design, and to obtain information on fuel capsule abalation by techniques similar to those used on RFD-1 for the reactor fuel rods. The instrumentation aboard the re-entry vehicle and at the ground station is similar to that used for RFD-1, but with increased emphasis on optical coverage. The re-entry vehicle has been designed with increased protection and better experi-

mental determination of the re-entry heat load distribution. It is expected to survive the re-entry and to be recovered from the Atlantic ocean about 200 miles southeast of Bermuda. Recovery will provide information about the burn-up of the nuclear power supply model and will provide useful information about the re-entry vehicle designe parameters.

In both demonstration flights the information obtained regarding the heating and burn-up of the nuclear power supply model is limited to disassembly information. No information has been or will be obtained from these flights regarding the final particle size and distribution in the upper atmosphere. This information would be an important input ot the aerospace nuclear safety program; however, as yet no practical method has been developed to obtain particle size and distribution during a flight test. Various schemes for sample collection aboard the re-entry vehicle have been examined and discarded as unworkable, mostly for aerodynamic reasons. Sampling by aircraft and rocket techniques is still under study with the hope that a practical system can be found. The possibility of tracking the debris cloud by radar or laser-radar techniques is being examined experimentally at Sandia Corporation. Knowledge of the dispersion of the debris cloud, as a function of time, could yield very useful information if a practical tracking scheme can be found. Until a method usable during a flight test can be found, we will remain dependent entirely on analytical and laboratory techniques for particle size and distribution information.

SUMMARY

SNAP systems appear promising for certain space power applications because of their reliability and long life. Steps are being taken by the USAEC to provide the radioisotope materials in sufficient quantities at prices that make the SNAP systems competitive with other power supplies. It is recognized that an adequate safety plan is an important part of the responsibility of the designer of each SNAP device. In addition, the Division of Reactor Development of the USAEC has established the office of Assistant Director for Nuclear Safety to perform an independent assessment of each safety plan. All of this effort is pointed toward providing assurance that the safety aspects of the SNAP applications are given thorough treatment.

The safety questions throughout the factory-to-launch sequence are readily treated and can be assessed by conventional analysis and testing techniques. Additional quantitative data regarding the conditions encountered during launch-pad accidents are being obtained by a coordinated DOD-NASA-USAEC program to provide better testing conditions for this circumstance. The main areas that occupy the attention of the aerospace nuclear safety organization are those associated with re-entry to the earth's atmosphere from suborbital and orbital flight. Although the controlled intact re-entry of the SNAP system into a designated recovery area would essentially remove all safety questions, this is impractical with present technology. Perhaps this possibility will open up as future developments in space technology occur; however, none of the SNAP systems that are expected to fly in the near future make use of intact re-entry. The designers of the SNAP systems to date have chosen re-entry burn-up and dispersal in the upper atmosphere as the mechanism for the disposal of the remaining radioactive material. If the SNAP device is located in the space vehicle so that it will be exposed to the effect of the re-entry heating, the stagnation heat available can be calculated readily. The work that has been done by the DOD and by NASA on re-entry heating has provided techniques for good heating calculations. To calculate the effect of this heat input on a particular SNAP device requires high-temperature materials properties information, for the materials involved. Since not all the information needed exists in good form, action is being taken to establish materials research programs that will complete the information. Very little is known about the stable ablation droplet size for various materials when they are heated to melting temperatures at high velocities in the upper atmosphere. As a part of the aerospace nuclear safety program, investigative work has been undertaken on this question, and promising techniques and apparatus are being developed which should yield quantitative information within the next year. A third major area of research is in the development of better analytical models to trace the fall and dispersion of particles of various sizes that are injected into the upper atmosphere. The successful completion of these research programs will provide a good capability for the analysis of the re-entry burn-up questions.

The need for a re-entry heating facility that would provide programmed re-entry conditions over long run times is evident. Such a facility could aid greatly in speeding the safety analysis of a given system and in assessing quickly the effect of design changes. A facility of this sort should be constructed as soon as a practical design can be developed.

Re-entry flight testing is an important (and expensive) part of the aerospace safety program. It is used to provide verification that calculations and laboratory testing are giving the correct picture of the re-entry heating effects. One such flight (RFD-1) was completed successfully in May 1963 to verify the disassembly features of a proposed SNAP reactor system, to check heating calculations on the full-scale model, and to obtain indications of reactor-fuel-rod ablation rates. A second flight is planned for the fall of 1964 to obtain similar information about a SNAP isotopic power supply model.

The safety of isotopic and reactor power supplies that are planned for space applications is considered thoroughly, first by the designer and second by the independent assessment of the Assistant Director for Nuclear Safety. The principal areas of uncertainty lie in materials properties, ablation droplet sizes, and upper-atmosphere fall and dispersion. Programs are underway to reduce these uncertainties so that the effectiveness of the re-entry burn-up disposal mechanism can be determined reliably for any SNAP system proposed for space application.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/286 Etats-Unis d'Amérique

Sûreté aérospatiale des sources d'énergie nucléaire

par G. A. Fowler

Le mémoire décrit le programme aérospatial de sûreté nucléaire exécuté par la Sandia Corporation pour le compte de la Commission de l'énergie atomique des Etats-Unis. Ce programme prévoit une évaluation indépendante de la sécurité d'emploi de l'énergie d'origine nucléaire dans les véhicules aérospatiaux. Il comprend l'établissement de modèles analytiques, des recherches, des essais au sol et des essais de rentrée dans l'atmosphère, toutes ces activités étant destinées à réduire les incertitudes quantitatives qui pourraient apparaître au cours de l'étude des mesures de sécurité à prévoir pour un système donné.

Un domaine de recherche qui intéresse la plupart des applications de l'énergie d'origine nucléaire pour alimenter les véhicules spatiaux est celui des phénomènes d'échauffement et de combustion qui se produisent lors du retour dans l'atmosphère. Au cours des dernières années, on a beaucoup étudié le problème de la résistance à l'échauffement qui se produit lors de la rentrée dans l'atmosphère, dans le cas de trajectoires relativement déclives comme celles de l'ICBM (Engin balistique intercontinental) ainsi que dans celui de trajectoires plus tendues tels les vols humains orbitaux. En revanche, l'utilisation délibérée de ce phénomène d'échauffement pour brûler et disperser diverses matières dans différentes configurations n'a présenté qu'un intérêt limité, sauf pour les constructeurs de générateurs aérospatiaux d'électricité nucléaire. Au cours des deux dernières années, la Sandia Corporation s'est notamment efforcée de mettre au point des méthodes expérimentales et analytiques permettant de calculer à l'avance l'échauffement et la combustion qui résultent du retour dans l'atmosphère. Ces recherches ont montré qu'il fallait améliorer les simulateurs et poursuivre les études sur les propriétés des matières. Lorsque l'échauffement est très élevé au moment du retour dans l'atmosphère, l'analyse a maintes fois prouvé que ce sont les propriétés des matières utilisées dans le générateur d'électricité que l'on connaît le plus mal. On s'efforce de combler les lacunes dans les connaissances sur les propriétés de ces matières. Les auteurs exposent le programme de recherches dans ce domaine.

Lorsqu'il y a combustion ou ablation par suite de l'échauffement produit par le retour dans l'atmosphère, la nature de la formation des gouttelettes et de la granulométrie finale des particules présentent alors de l'intérêt en ce qui concerne les matières à l'étude. Les auteurs examinent les résultats des recherches sur le mécanisme de la formation de gouttelettes lors de l'ablation. Ils précisent où en sont les connaissances relatives au calcul théorique du transport, de la dispersion et du dépôt des particules introduites dans la haute atmosphère du fait de la combustion qui se produit lors du retour.

L'évaluation de la sûreté d'un générateur nucléaire d'électricité pour véhicule aérospatial, dans toutes les conditions de manipulation et de transport, depuis la construction jusqu'au vol sous-orbital, exige divers essais au sol pour simuler les conditions auxquelles le générateur pourrait être soumis. On a réussi à simuler efficacement les conditions d'impact dans divers

milieux à l'aide d'installations appropriées. Etant donné le caractère limité des données quantitatives connues sur les niveaux de surpression et sur les conditions température-temps lors d'explosions et d'incendies sur les plates-formes de lancement, on a décidé, pour y remédier, de doter d'instruments plusieurs aires de lancement de manière à obtenir des données précises afin de réaliser une meilleure simulation lors des essais au sol.

Pour vérifier les caractéristiques de « désassemblage » données par le constructeur à un dispositif nucléaire aérospatial et pour éprouver les méthodes actuelles de prévision de l'échauffement dû à la rentrée dans l'atmosphère, on a procédé en mai 1963, dans la région de l'île Wallops, à un essai en vol à l'aide d'un engin Scout de la NASA (Administration américaine pour l'aéronautique et l'espace) chargé de faire rentrer dans l'atmosphère, selon une trajectoire tendue, un satellite de la Sandia Corporation. On a obtenu de bons renseignements par télémesure jusqu'à la suppression du faisceau hertzien par les phénomènes d'ionisation, et des données optiques ont été recueillies par des caméras au sol et par des avions. Les auteurs rendent brièvement compte de cet essai en vol et des résultats obtenus, et décrivent succinctement le deuxième essai de rentrée dans l'atmosphère prévu pour 1964.

A/286 США

Обеспечение радиационной безопасности при запусках в космос изотопных и реакторных источников

Г. А Фаулер

В докладе описывается программа обеспечения радиационной безопасности при полетах в космос, проводимая фирмой «Сандиа корпорейшен» для Коммиссии по атомной энергии США. Программа предусматривает получение объективной оценки радиационной опасности при использовании ядерной энергии на космических кораблях. В программу включены изучение аналитических моделей, научно-исследовательская работа, наземные испытания и испытания в полете с возвращением в атмосферу. Она направлена на увеличение точности оценки систем радиационной безопасности конкретных установок.

Одна из областей исследования, относящаяся к использованию ядерных источников питания на космических кораблях, касается нагрева и абляции при входе в плотные слои атмосферы. Большое внимание уделяется вопросу устойчивости к нагреву при спуске как при сравнительно крутых траекториях возврата межконтинентального баллистического снаряда, так и при более пологих траекториях возврата пилотируемого аппарата. С другой стороны, использование нагрева при орбитальном полете для сжигания и рассеивания различных материалов, имеющих различные конфигурации, вызывало незначительный интерес всех разработчиков, помимо конструкторов космических ядерных источников энергии. Исследования, проведенные фирмой «Сандиа корпорейшен» за последние два года, включали изучение экспериментальных и аналитических методов определения возможного нагрева и сгорания при входе в плотные слои атмосферы. Эти работы показали необходимость создания усовершенствованных моделирующих устройств и проведения дальнейших исследований свойств материалов. Если возврат в плотные слои атмосферы осуществляется при высоких температурах, свойства материалов, используемых в ядерном двигателе, являются как правило, одним из главных источников ошибок аналитических расчетов. Принимаются меры для заполнения пробелов в изучении свойств материалов, применяемых в источниках энергии космических аппаратов. Приводится описание программы исследований свойств материалов.

Если достигнуты условия оплавления или абляции во время нагрева корабля при входе в плотные слои атмосферы, природа образования капель и конечное распределение частиц по размеру представляют большой интерес. Обсуждаются успехи, достигнутые в изучении механизма абляционного образования капель. Описывается современное состояние знаний в области предсказаний переноса, дисперсии и осаждения частиц, введенных в верхние слои атмосферы при абляции, обусловленной входом корабля в плотные слои атмосферы.

Оценка безопасности космической ядерной силовой установки на всех стадиях, от изготовления до суборбитального полета, требует проведения различных наземных испытаний, моделирующих условия, с которыми может столкнуться ядерная установка. Условия ударной нагрузки в различных средах успешно осуществлялись различными методами, основанными на применении ракетных салазок. Недостаточность количественных данных об уровнях избыточных давлений и изменении температуры во времени была обусловлена взрывами на пусковых столах и пожарами, в результате чего было решено оборудовать несколько пусковых площадок с целью сбора данных этого типа, при наличии которых можно достигнуть более точного моделирования при проведении наземных испытаний.

Для проверки демонтажных характеристик, заложенных конструктором в космическую ядерную установку, а также для выяснения имеющихся в настоящее время возможностей предсказания нагрева при входе в плотные слои атмосферы в мае 1963 года были проведены летные испытания. Для выведения кабины с по-

лезной нагрузкой на орбиту с пологой траекторией возврата в плотные слои атмосферы использовалась ракета «Скаут» Национального управления по аэронавтике и исследованию космического пространства. Запуск производился с полигона в районе острова Уоллос. До возникновения ионизационного запирания были получены хорошие телеметрические данные. Оптические данные были получены от надежных камер и с самолета. Дается краткое изложение результатов проведенных летных испытаний и коротко описывается программа второго полета с возвратом в плотные слои атмосферы, намеченного на 1964 год.

A/286 Estados Unidos de América

La seguridad aeroespacial de fuentes de energía constituidas por isótopos o reactores

por G. A. Fowler

Esta memoria describe el programa de seguridad nuclear aeroespacial que está llevando a cabo la Sandia Corporation para la Atomic Energy Commission de los Estados Unidos. El programa tiene por objeto una evaluación independiente de los aspectos de seguridad relacionados con la aplicación de fuentes de energía nucleares a los vehículos espaciales. El programa de seguridad incluye modelos analíticos, investigaciones, ensayos en tierra y en vuelo de regreso, dirigidos a reducir el margen de incertidumbre que puede presentarse al revisar el plan de seguridad de un sistema determinado.

Un campo importante de investigación en la mayoría de las aplicaciones espaciales de fuentes de energía nuclear es el del calentamiento en el regreso a la atmósfera y fenómenos de combustión. El problema de la superación del calentamiento de regreso ha recibido una atención considerable en los últimos años, tanto para las trayectorias de regreso relativamente brusco de los ICBM como para los descensos más suaves en los vuelos orbitales pilotados. Por otra parte, ha sido de limitado interés, excepto para los proyectistas de fuentes espaciales de energía de origen nuclear, la utilización deliberada de los mecanismos de calentamiento en la fase de regreso orbital con objeto de conseguir la combustión y dispersión de varios materiales en configuraciones diversas. El esfuerzo realizado por la Sandia Corporation en los dos últimos años incluye la investigación de técnicas analíticas y experimentales para la predicción del calentamiento y combustión en el regreso. Estas investigaciones han resaltado la necesidad de mejorar las instalaciones de simulación y de realizar estudios adicionales sobre las propiedades de los materiales. En condiciones de regreso correspondientes a elevadas temperaturas, una de las principales causas de indeterminación de muchos análisis parece estar constituida por las propiedades de los materiales utilizados para las fuentes de energía. Se están realizando los trabajos adecuados para llenar las lagunas que existen en el conocimiento de esas propiedades de los materiales utilizados en los sistemas generadores de energía de tipo nuclear con aplicaciones aeroespaciales. Se expone un esquema del programa de investigación.

Una vez alcanzadas las condiciones de combustión o desgates en el calentamiento de regreso, es interesante conocer para cada uno de los materiales estudiados, las condiciones del proceso de formación de gotitas y la distribución final del tamaño de partículas. Se discute el progreso realizado en las investigaciones sobre los mecanismos de formación de gotitas por el desgaste. Se describe el estado actual de conocimientos sobre la predicción del transporte, dispersión y deposición de partículas introducidas en las capas altas de la atmósfera por la combustión durante el regreso.

La evaluación de la seguridad de las fuentes de energía de origen nuclear aeroespaciales en todas las condiciones de manejo y transporte, desde la fabricación al vuelo suborbital, exige diversas posibilidades de ensayo en tierra para simular las condiciones a las que puede estar sometido el generador nuclear. Se han simulado eficazmente las condiciones de impacto en varios medios mediante diversas técnicas con arrastre por cohetes. El limitado conocimiento cuantitativo que existe sobre los niveles de sobrepresión e historias temperatura-tiempo causados por las explosiones y los fuegos durante el lanzamiento, ha llevado a la confección de un programa que, al dotar de instrumentación adecuada diversos puntos del lugar del lanzamiento, permitirá recoger esta clase de información. De este modo podrá conseguirse una simulación más adecuada en los ensayos realizados en tierra.

Para comprobar las características de desmantelamiento incorporadas por el proyectista en un determinado sistema nuclear espacial, así como para determinar la posibilidad que existe actualmente de predecir el calentamiento de regreso, se realizó, en mayo de 1963, un vuelo de ensayo utilizando un vehículo explorador NASA; se colocó una carga útil tipo Sandia en un vuelo de regreso de poca altura sobre la isla de Wallops. Se obtuvieron buenos datos por telemedición hasta que se produjo oscurecimiento por ionización y se obtuvieron también datos ópticos mediante cámaras fotográficas instaladas en tierra y en aviones. Se presenta un resumen de este vuelo de ensayo y sus resultados, así como una breve descripción del segundo vuelo de regreso planeado para 1964.

Gamma energy release rates from fission products at short times after fission

by M. Ramakrishna Iyer and A. K. Ganguly *

CALCULATION OF GAMMA ENERGY RELEASE RATES FROM FISSION PRODUCTS AT SHORT TIMES AFTER FISSION

The gamma energy release rates from fission product mixtures have been theoretically estimated by various workers [1–3]. Many of these attempts are limited to post-fission times of a few minutes and longer. This is mainly because of the limitation of input data. The present work has attempted to improve existing values of gamma energy release rates by including information on more fission products—mainly short-lived ones. With this object in view, a compilation [4] of data on the nuclear properties of uranium-235 fission products (complete information available up to June 1962) has been made. 315 fission products both stable and unstable whose existence is known or suspected are included in the compilation. Due to the absence of experimental information, the gamma decay energies of 22 short-lived fission products were theoretically computed using beta decay theory and the nuclear shell model. A ten-group classification of fission product gamma energy has been adopted in the present work. The groupings have been derived from the point of view of dose computation. Gamma energy release rates, photon emission rates and weighted average energy have been studied in each group and the suitability of the groupings is discussed.

Calculation

Calculations were carried out for each fission product decay chain using the Bateman solution taking into account the contribution of all members of the chain. The gamma photons emitted by fission products were divided into the following ten energy groups:

Group No.	Group limit (MeV)
1	0 –0.275
2	0.275–0.55
3	0.55 –0.825
4	0.825–1.1
5	1.1 –1.38
6	1.38 –1.65
7	1.65 –2.00
8	2.00 –2.5
9	2.5 –3.00
10	> 3.00

The photon emission rates and gamma energy release rates in each group were calculated by summing up the release by each fission product in that particular group. Expressions for photon emission rate, gamma energy release rate and weighted average energy for the gth group (g = 1 to 10) are respectively:

$$\sum_i N_i(t)\,\lambda_i\,p_{g,i}$$

$$\sum_i N_i(t)\,\lambda_i\,p_{g,i}\,E_{g,i}$$

and

$$\frac{\sum_i N_i(t)\,\lambda_i\,p_{g,i}\,E_{g,i}}{\sum_i N_i(t)\,\lambda_i\,p_{g,i}}$$

where $N_i(t)\,\lambda_i$ is the activity of the ith product at time t and $p_{g,i}$ is the photon yield per disintegration of ith product having energies in the gth group i.e. $E_{g,i}$. Out of the 315 fission products given in the compilation [4], 77 are stable nuclides. Of the 238 radioactive fission products, 53 are not included in the present computations because of the lack of data. The remaining 185 products, including the 22 for which the gamma spectrum was estimated theoretically, are considered here.

Results

Gamma energy release rates in the ten energy groups are plotted as a function of cooling time in Figs. 1 and 2. Total gamma energy release rates and photon emission rates are also given in Fig. 2. Figure 3 gives the weighted average energy for each group as well as for total fission products. In Fig. 4 the total gamma energy

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release rate obtained in the present work is compared with those of Way and Wigner [5], Perkins and King [3] and Zigman and Mackin [6].

The computed energy spectrum for short cooling times is not expected to be entirely accurate due to the uncertainity of the gamma spectra of very short-lived fission products. The spectral estimates for the short-lived fission products are approximate but the estimates of total gamma energy release from these are expected to be reasonable. From 30 to 500 s after fission the gamma energy release in group 10 predominates, mainly due to the high energy gammas of ^{87}Br ($t_{\frac{1}{2}} =$ 55.6 s) and ^{136}I. ($t_{\frac{1}{2}} = 82$ s). After 500 s this group decays rapidly. The next important group in the same time region is group 5. This is due to ^{136}I (1.32 MeV gamma). The contribution of group 3 to the total gamma energy release is equally important throughout the entire range of time considered. Other groups, except 1 and 9, contribute more or less in equal measure to the total energy release rate. The contribution of group 1 to the total gamma energy release rate is found to be small almost throughout the decay time considered and the contribution of group 9 becomes small after about 2×10^3 s.

The maximum deviation of the weighted average energy from its mid-value in the groups is small for all groups ($< 9.6\%$) except 1 and 2, for which it is about 20%. The short-lived fission products for which gamma spectra were theoretically estimated have yielded only low energies, while some of the longer lived ones like ^{87}Br and ^{136}I have harder gammas. Activity of the former at short times over shadow ^{87}Br and ^{136}I while these become significant at about 100 s. Thus weighted average energy of the total fission products increases with time in the time region of 100 s.

Grouping of gamma energies

From the point of view of dose computations, the groupings should be such that the effective $E \times \mu_a(E)$ for fission product gammas in a group does not vary appreciably with time. The weighted average $E \times \mu_a(E)$ for the g^{th} group is given by:

$$\frac{\sum_i N_i(t)\,\lambda_i\,p_{g,i} \times E_{g,i}\,\mu_a(E_{g,i})}{\sum_i N_i(t)\,\lambda_i\,p_{g,i}}$$

where $\mu_a(E_{g,i})$ is the linear energy absorption coefficient in air of photons of energy $E_{g,i}$. The maximum deviation of the weighted average $E \times \mu_a(E)$ from its mid-value is less than 6% for all groups except 1 and 2. For groups 1 and 2, this is about 17%. However, the contribution of groups 1 and 2 to the total energy release does not exceed 13% up to 10^4 s and at 10^5 s it is 25%. Hence a maximum error of 5% only is reflected in the computation of total dose from these groups (by assuming constant $E \times \mu_a(E)$).

Conclusions

Gamma energy values obtained in the present work are compared with the theoretical and experimental results of earlier workers in Fig. 4. Gamma energy release rates computed here agree better with the experimental values than the calculated values of Perkins and King [3] at times of less than 1 000 s of cooling. The present values are higher than those of Way and Wigner [5] at decay times more than 90 s. The same situation is obtained for the results of Perkins and King at about 400 s. At times greater than 1 000 s the present values are in good agreement with the values of Perkins and King. At times less than 100 s agreement between the calculated values of the present work and the experimental values of Zigman and Mackin [6] is poor. At short times of cooling only experimental measurements can yield reliable results.

The weighted average energy of the gammas is found to increase from 0.9 MeV/photon at 0.1 s of cooling to about 1.40 MeV/photon at about 2×10^2 s and thereafter decreases gradually to about 0.53 MeV/photon at about 10^5 s. The values of photons per disintegration are calculated at different times of cooling from the average gamma energy per disintegration of Perkins and King and the average gamma energy per photon of the present work. It can be stated that to a fair approximation one photon per disintegration is emitted from fission products in the time region of 10^2 to 10^5 s. However, as the present computation includes more recent data on a number of short-lived fission products which are not included by Perkins and King, the photons per disintegration obtained at times less than 10^3 s may be fortuitous.

EXPERIMENTAL INVESTIGATION OF GAMMA ENERGY RELEASE RATES FROM FISSION PRODUCTS AT SHORT TIMES AFTER FISSION

Way and Wigner [7] theoretically predict the total energy (neutrinos, beta and gamma) release from fission products from 10^{-4} s to 10^8 s of cooling. It can be stated that approximately one-fourth of the total energy is given out by gamma photons. The total energy vs. time curve of Way and Wigner flattens at times of cooling less than 10^{-2} s and approaches 3.8 MeV/fission s. This indicates the existence of a lower limit for the half-life of fission products decaying by beta emission. The present experimental programme is aimed at finding the decay rate of fission product gammas at short times of cooling, especially in the unexplored time regions from a few milliseconds to about one second.

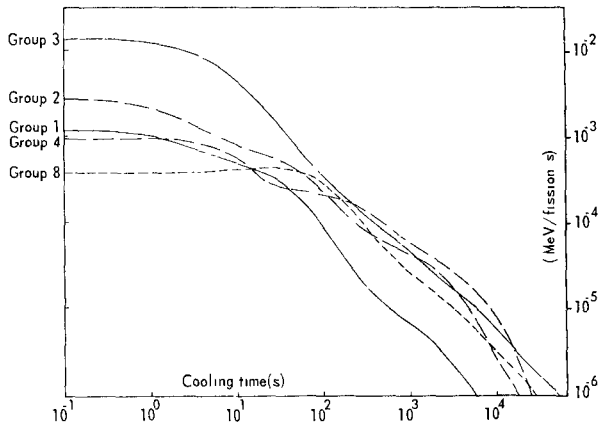


Figure 1. Calculated gamma energy release from fission products

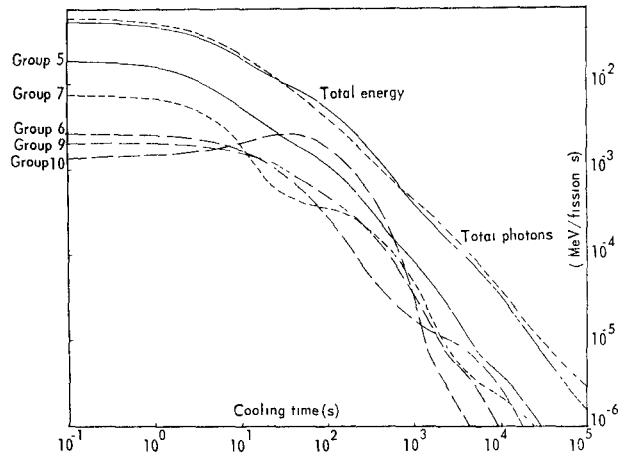


Figure 2. Calculated gamma energy release from fission products

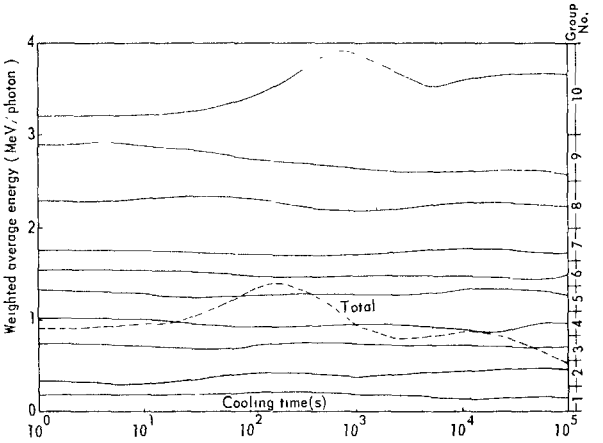


Figure 3. Weighted average energy for the groups and for total fission products

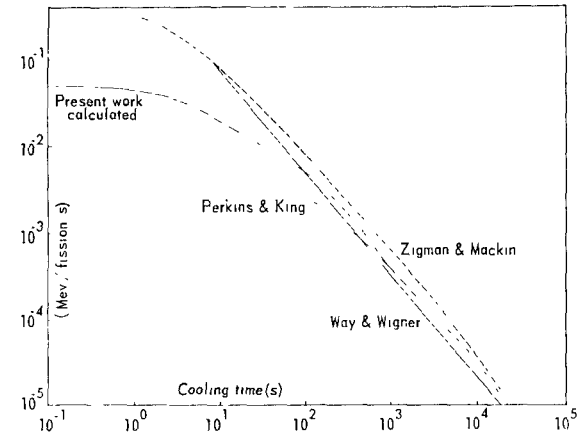


Figure 4. Total gamma energy release rates from fission products: comparison with the results of earlier workers

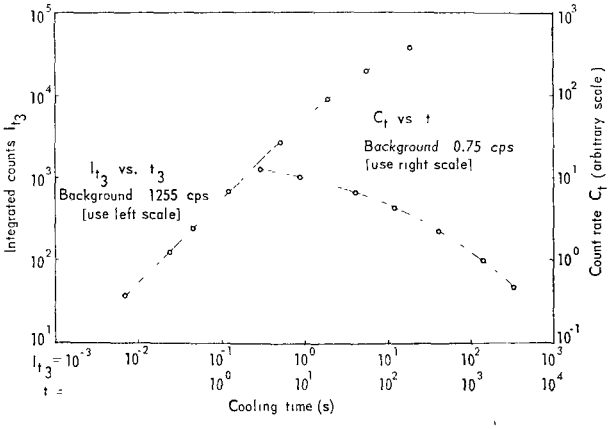


Figure 5. Movie film I_{t3} values and recorder counts above background (C_t)

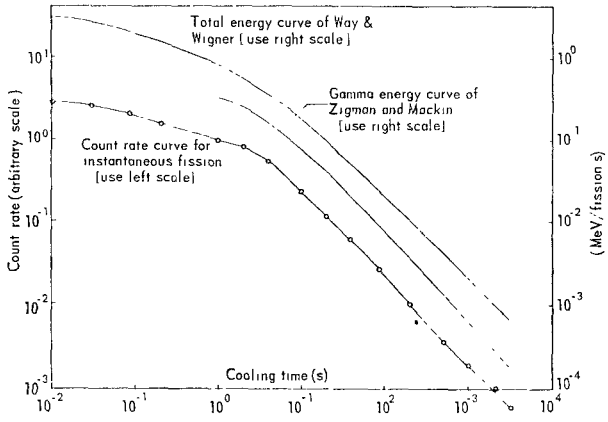


Figure 6. Count rate curves for instantaneous fission

Experimental technique

The technique consisted of irradiating a sample of 95% enriched uranium in a capsule and bringing it pneumatically to the detector within a short time. The tube in which the sample capsule moves was a short one (37 cm). Contact switches fixed to the ends of the tube operated when the capsule was at either end. One end of the tube faced a collimated beam (thermal neutron flux 3.2×10^6 n/cm^2 s) and the other end was in contact with a 1 in $\times$ 1 in NaI (Tl) crystal. The separation of the sample from the detector surface was nearly 0.5 cm.

The time of irradiation and time of travel of the capsule (from the beam to the detector) were measured by means of three scalers (synchronised within 2 in

300 000 counts) counting a standard pulse rate of 1 000 cps. By means of relay arrangements and the contact switches, the scalers Sc_1 and Sc_2 started counting at the start of the sample irradiation, Sc_2 alone being stopped at the end of irradiation. The third scaler Sc_3 started when the sample reached the detector. Sc_1 and Sc_3 were stopped simultaneously after the experiment. Scaler Sc_4 started counting the output from the detector as soon as the sample reached the detector. $Sc_2/1\,000$ and $[Sc_1 - (Sc_2 + Sc_3)]/1\,000$ give the irradiation time and travel time of the sample respectively.

For following the record of counts, simultaneous fast movie photos of the counting panel of the scalers Sc_1 and Sc_4 were taken. The speed of the camera was maintained kept at 24 shots per second and the exposure time at approximately 1/250 of a second. The camera was started just before the capsule arrived at the detector and ran for about 20 s. The camera recorded the position of the neon dots and register readings of the scalers.

A dummy capsule without any uranium was irradiated for about 1 000 s and counted at the detector. No increase in the integrated counts up to times of the order of one second was found with the irradiated sample. In one of the experiments, the irradiation time t_1 and travel time of the uranium sample were found to be 6.959 s and 0.008 s respectively. The integrated counts I_{t_3} up to different times after the arrival of the sample at the detector as read from the movie film are given in Fig. 5.

Converting the results for instantaneous fission

Let the decay law of count rate for instantaneous fission be $\Phi(t)$ and let R be the fission rate. The integrated counts I_{t_3} up to t_3 seconds after the arrival of the capsule at the detector is:

$$I_{t_3} = \int_{t=0}^{t_1} R\,dt \int_{t'=0}^{t_3} \Phi(t_1 - t + t_2 + t')\,dt'$$

If the results of I_{t_3} vs. t_3 are fitted to the above form then $\Phi(t)$ (for $t > t_2$ in the region of the fit) can be determined. The correct and most general form of $\Phi(t)$ would be $\sum_i A_i e^{-a_i t}$. A function of two exponential terms was found to be adequate for obtaining a good fit. The fittings were performed in two time regions, i.e., t_3 from 0 to 2 s and 2 to 20 s. Table 1 gives the fitted function, fitting errors and the form of $\Phi(t)$ obtained from the fit.

Experiment for longer cooling times

A recorder was used to follow the decay of the irradiated sample in the range 3 to 3×10^3 s after fission. As the time of travel was no longer of serious concern the experiment was carried out at a higher flux position. The time of irradiation and time of travel in the experiment were respectively 898 and 1.18 s. The count rate vs. time of cooling is presented in Fig. 5. Using a function of eight exponential terms, it was possible to obtain a good fit (maximum error of the fit being less than 3.8%) for the time range of 3 to 3×10^3 s. $\Phi(t)$, the decay law for instantaneous fission, was obtained from the fit. Results of the two experiments were normalised at 10 s and the count rate for instantaneous fission thus obtained is given in Fig. 6.

Gamma energy release rates

The efficiency of a sodium iodide crystal varies with the energy of photon. With the known data of fission products presented in the first part of the paper and the values of efficiency given by Wollicki *et al.* [8], the effective efficiency for the detector system used was calculated for different times of cooling. The effective efficiency is found to vary within a small range of 0.092 to 0.098 during the cooling time of 10^{-1} to 10^3 s. The gamma energy of photons from short-lived fission products do not fall outside the range of photon energy considered for the above computation. Hence we conclude that the counts for the total fission products follow the decay of the gammas themselves in the entire time range considered.

The decay of the count rate curve is plotted along with the total energy release rate curve of Way and Wigner [7] and the experimental gamma energy release curve of Zigman and Mackin [6] in Fig. 6. Ratios of count rate to the total energy values of Way and Wigner (from 10^{-2} to 3×10^3 s) and to the gamma energy values of Zigman and Mackin (from 1 to 3×10^3 s) are found to be reasonably constant with time, the maximum deviations being 22.4 and 10.3% respectively. The average value (3.00) of the ratios of count rate to the gamma energy release rate from 1 to 3×10^3 s is used to convert the count rates of the present experiment to gamma energy release rates (*cf.* Fig. 7). The

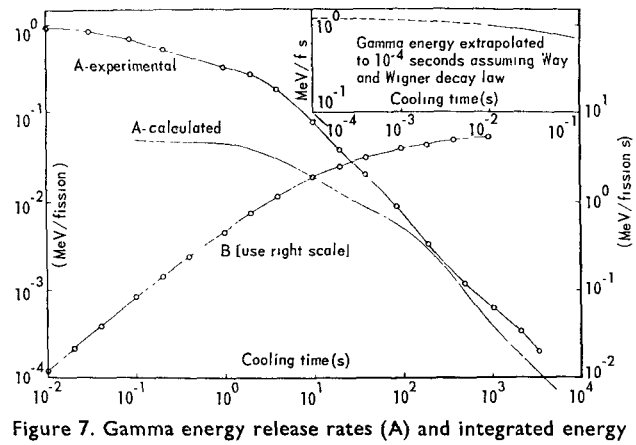


Figure 7. Gamma energy release rates (A) and integrated energy release (B)

Table 1. Fitting the observed readings of I_{t3} vs. t_3

(*a*) $t_3 = 0$ to 2 s

$$I_{t_3} = 34.78\,(1 - e^{-7.16695\,t_3}) + 26\,537.12\,(1 - e^{-0.19857\,t_3})$$

$$\Phi(t) = 1\,891.96\,e^{-7.16695\,t} + 1\,399.46\,e^{-0.19587\,t}$$

Fitting errors

t_3 (s)	Actual I_{t_3}	Fitted I_{t_3}	Fitting error (%)
0.007	36	38.6	+ 7.2
0.047	241	256.5	+ 6.4
0.206	1 086	1 090.4	+ 0.4
0.406	2 085	2 088.3	+ 0.15
1.006	4 832	4 839.8	+ 0.16
2.005	8 759	8 750.1	— 0.10

(*b*) $t_3 = 2$ to 20 s

$$I_{t_3} = 8\,241.56\,(1 - e^{-0.36527\,t_3}) + 41\,713.37\,(1 - e^{-0.05801\,t_3})$$

$$\Phi(t) = 1\,197.08\,e^{-0.36527\,t} + 422.79\,e^{-0.5801\,t}$$

Fitting errors

t_3 (s)	Actual I_{t_3}	Fitted I_{t_3}	Fitting error (%)
2.005	8 759	8 860	+ 1.14
4.009	14 864	14 992	+ 0.86
6.014	19 410	19 611	+ 1.03
8.013	23 812	23 308	— 2.11
10.013	26 287	26 407	+ 0.45
20.020	36 896	36 891	— 0.01

graph inset in Fig. 7 shows the present experimental curve of gamma energy release rates extrapolated (dashed line) 3 to 10^{-4} s assuming the same pattern of decay as the total energy curve of Way and Wigner.

Discussion

The experimental gamma energy release rate curves tend to flatten at short times of cooling. This agrees with the observation of Way and Wigner. Fermi and King as quoted by Way and Wigner [7] found the gamma energy emission rate to be roughly constant from 40 to 140 ms after fission. The present curve has a slope in this region. Carver [9] has extrapolated arbitrarily the gamma energy release from fission products by bridging the energy release arising from the beta decay of fission fragments (for times greater than one second) and the prompt gamma energy release arising from the excited states of fission fragments (for times less than one microsecond) obtained experimentally by Maienschien *et al.* [10]. In the region of 10 ms the slope of the extrapolated curve is higher than that obtained in the present work.

For a comparative study of the gamma energy released from fission products up to various times of cooling, the total gamma energy released in MeV/fission is obtained by integrating the experimental gamma energy release rate curve from 10 ms up to various times of cooling. The gamma energy release up to 10 ms is obtained from the extrapolated curve of Fig. 7 and the upper limit of the energy release in this region is found to be 1.2×10^{-2} MeV/fission. This is added to the total gamma energy release values obtained in the experiment, and the resultant curve is presented in Fig. 7. Integrated gamma energy release up to one second after fission is found to be 0.454 MeV/fission. The total gamma energy released beyond one second of cooling was found to be 6.6 MeV/fission by Knabe and Putnam [11]. Thus the total gamma energy release from fission products comes to 7.054 MeV/fission. The per cent of gamma energy released from 10^{-2} to 1 s is 97% of the total gamma energy release up to one second. On the basis of the extrapolated curve, Carver found that 95% of the total gamma energy released up to one second comes from 10^{-3} to 1 s. However, the energy released up to one second, 0.679 MeV/fission, suggested by Carver is higher than that obtained in the present work.

ACKNOWLEDGEMENT

Our thanks are due to Shri S. D. Soman and Shri D. V. Gopinath for helpful discussions.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/792 Inde

Taux de libération de l'énergie gamma des produits de fission pour des temps courts de désactivation

par M. Ramakrishna Iyer et A. K. Ganguly

On a évalué les taux de libération d'énergie gamma et les taux d'émission de photons à partir d'un mélange de produits de fission, formé dans la fission instantanée de l'uranium 235. Nous avons utilisé les données de notre dernier recueil (complet jusqu'à juin 1962) sur les propriétés nucléaires des produits de fission de l'uranium 235 pour les rendements de fission, les demi-vies et les énergies de désintégration. Le spectre gamma est présenté en divisant les énergies gamma des produits de fission en dix groupes d'après les énergies moyennes corrigées et les caractéristiques de la dose moyenne corrigée dans l'air. La distribution des taux de libération d'énergie gamma, les taux d'émission de photons et l'énergie moyenne corrigée dans chaque groupe et dans l'ensemble des produits de fission ont été étudiés. On représente les résultats pour des vies comprises entre 10^{-1} et 10^{5} s. On trouve que les taux de libération d'énergie gamma présentés dans le mémoire sont nettement meilleurs que les valeurs estimées données jusqu'ici pour des temps de désactivation de moins de 1 000 s. Au-dessus de 1 000 s, les valeurs sont en accord avec les valeurs estimées obtenues par les chercheurs précédents.

Une nouvelle technique expérimentale a été mise au point pour estimer en fonction du temps les taux de libération d'énergie gamma, pour des temps de désactivation de l'ordre de la milliseconde et plus. La décroissance des rayons gamma à partir de cet intervalle de temps a été suivie en prenant des photos cinématographiques rapides des démultiplicateurs; on a adopté la méthode qui consiste à compter les photons des produits de fission pour convertir les résultats obtenus pour l'irradiation finie au cas de la fission instantanée. La courbe expérimentale du taux de comptage en fonction du temps a une pente légèrement plus forte que la courbe expérimentale de Fermi et King et elle est plus ou moins en accord avec la pente de la courbe d'énergie totale de Way et Wigner. La conjecture de Carver sur la vitesse de libération d'énергie gamma pour des temps d'une fraction de seconde après la fission donne une pente plus forte que les résultats obtenus dans l'expérience décrite.

A/792 Индия

Скорость выделения энергии γ-излучения из продуктов деления при небольшом времени выдержки

М. Рамакришна Иер, А. К. Гангули

В докладе рассчитываются скорости выделения энергии γ-излучения и скорости испускания фотонов из смеси продуктов деления, образующихся при мгновенном делении U^{235}. Данные по выходам деления, периодам полураспада и энергиям γ-излучения заимствованы из нашего последнего атласа (составленного на июнь 1962 года) ядерных свойств продуктов деления U^{235}. Представлены γ-спектры путем распределения энергии γ-излучения продуктов деления на десять групп на основе их средневзвешенных энергий и средневзвешенных характеристик дозы в воздухе. Исследовалось изменение скоростей выделения энергии γ-излучения, скоростей испускания фотонов и средневзвешенной энергии в каждой группе и для всех продуктов деления. Результаты построены

в виде графиков для времен распада от 10^{-1} до 10^5 *сек*. Установлено, что представленные в докладе данные по скорости выделения энергии γ-излучения значительно превосходят существующие вычисленные значения при времени выдержки менее 1000 *сек*. Для времени выдержки более 1000 *сек* значения согласуются со значениями, вычисленными ранее другими авторами.

Разработана новая экспериментальная методика оценки временной зависимости скорости выделения энергии γ-излучения при времени выдержки в области миллисекунд и выше. Начиная с этого времени γ-распад прослеживали при помощи высокоскоростного фотографирования и приборов, регистрирующих фотоны, вылетающие из продуктов деления, причем эти данные были приспособлены для преобразования результатов, полученных для конечного облучения, к случаю мгновенного деления. Экспериментальная кривая зависимости скорости счета от времени несколько круче, чем экспериментальная кривая Ферми и Кинга, и более или менее согласуется с наклоном кривой полной энергии Уэй и Вигнера. Представления Карвера о скорости выделения энергии γ-излучения в области времени долей секунды после деления дают более крутой наклон, чем результаты, полученные в настоящем эксперименте.

A/792 India

Liberación de energía gamma por los productos de fisión al cabo de breves períodos de enfriamiento

por M. Ramakrishna Iyer y A. K. Ganguly

En la memoria se calcula el índice de liberación de energía gamma y de emisión de fotones por una mezcla de productos de fisión resultante de la fisión instantánea del uranio-235. Los datos sobre rendimiento de fisión, períodos de semidesintegración y energía de desintegración gamma proceden de nuestra última compilación (completa hasta junio de 1962) de las propiedades nucleares de los productos de fisión del uranio-235. El espectro gamma se define dividiendo las energías gamma de los productos de fisión en diez grupos determinados sobre la base de sus respectivas energías medias ponderadas y de las características de la dosis media ponderada en aire. Se ha estudiado la variación de la cantidad de energía gamma liberada, de la cantidad de fotones emitidos y de la energía media ponderada en cada grupo y en los productos globales de fisión. Los resultados se representan gráficamente para tiempos de desintegración comprendidos entre 10^{-1} y 10^5 s. Se considera que las cantidades de energía gamma liberada que figuran en la memoria constituyen un progreso apreciable respecto de los valores utilizados hasta el presente, para tiempos de enfriamiento inferiores a 1 000 s. Por encima de ese tiempo, los valores concuerdan con los calculados anteriormente por otros investigadores.

Se ha ideado una nueva técnica experimental para calcular la variación, en función del tiempo, de la cantidad de energía gamma liberada al cabo de tiempos de enfriamiento del orden del milisegundo o mayores. La disminución de la actividad gamma a partir de este intervalo de tiempo se ha observado fotografiando los escalímetros con una cámara cinematográfica rápida y se ha recurrido al recuento de los fotones emitidos por los productos de fisión para transformar los resultados relativos a la irradiación finita de modo que sean aplicables al caso de la fisión instantánea. La pendiente de la curva experimental que representa el índice de recuento en función del tiempo es ligeramente mayor que la de la curva experimental de Fermi y King, y coincide aproximadamente con la pendiente de la curva de energía total de Way y Wigner. De las teorías de Garver acerca de la cantidad de energía gamma liberada en un intervalo de tiempo del orden de una fracción de segundo después de la fisión, se deduce una pendiente más acusada que de los resultados obtenidos en el presente experimento.

Record of session 3.9

Nuclear safety research

Chairman: F. R. Farmer (United Kingdom)

Reactor kinetics and transient studies

Paper P/282 (presented by J. A. Lieberman)

DISCUSSION *

B. G. DUBOVSKY (USSR): In designing fast reactors, do you take into account the possibility of a positive reactivity effect (due to accidental displacement of sodium coolant and resultant hardening of the neutron spectrum) occurring together with defects in the safety system? What engineering safeguards do you include in the design?

J. A. LIEBERMAN (United States of America): I think Mr. McCarthy would be the one to answer that.

W. J. MCCARTHY (United States of America): The possibility of a positive Doppler coefficient due to loss of sodium was first pointed out by Okrent and Dickerman. This possibility has not been specifically taken into account in the design of the 1 000 MW(e) reactors, other than to provide the usual stringent precautions against loss of sodium either generally or locally. In any serious design, this possibility would need to be considered in detail.

B. G. DUBOVSKY (USSR): On the basis of your experience so far, is it possible to evaluate the expediency of using large protective shells in reactors, so as to prevent the spread of radioactive contamination in the event of an accident?

J. A. LIEBERMAN (United States of America): A number of containment vessels for water reactor plants have been successfully built and operated in the United States. Detailed leak rate tests have been carried out and daily leak rates of 0.1 % of the contained volume or less have been achieved. I would refer you to a handbook on containment to be put out by Cottrell *et al.* of Oak Ridge National Laboratory, which should be of interest to you in this regard.

W. HÄFELE (Federal Republic of Germany): Release of air to the atmosphere from most reactor containment buildings is effected through sets of filters. Normally these filters will stand only a slight change in pressure. Is there any work going on in the United States to develop pressure-resistant high-temperature filters? What are the pressure and temperature limits of the best filters available? What are the corresponding filter factors?

J. A. LIEBERMAN (United States of America): This is an important question. One should know how well an air-cleaning system will perform under accident conditions, i.e., abnormal temperature and pressure. The absolute air filters (pleated-paper type) in current use can withstand pressures only in the range of several inches of water and a few hundred degrees Fahrenheit. One of the important objectives of the work described in our paper is to obtain engineering data on such system performance under conditions as similar as possible to those that would prevail following a major accident.

The air-cleaning devices generally used provide removal efficiencies of 99.8 % or better for particles 0.3 μ in diameter and about 99.95 % for iodine, under normal operating conditions.

K. A. MAHMOUD (United Arab Republic): I understand from the paper that your budget for the work on nuclear safety is $18 million for the year 1964. What percentage does this represent of the annual budget allotted to work on the peaceful uses of atomic energy in the United States? I would also like to know what percentage of the personnel is engaged on problems of nuclear safety.

J. A. LIEBERMAN (United States of America): While I do not have definite figures or percentages available, the percentage of the total budget devoted to development of nuclear safety research is quite small, probably about 4 %. However, this does not include safety work that is carried out in connection with specific reactor projects; nor work related to health and environmental surveillance activities.

The estimated number of people involved in the programme described is 400–500.

R. D. COLLINS (United Kingdom): ** In the section on fission-product behaviour in paper P/282, it is stated that under some conditions iodine tends to react with

* See also General Discussion in this session.

** This comment was not read out during the meeting.

impurities to form compounds which are not effectively removed by current air-cleaning systems. It is true that compounds can be formed which are harder to trap than iodine vapour and, as reported in paper P/186, we have identified some of these, methyl iodide being the most important. But over a considerable number of experiments chosen to include all the conditions relevant to accidents postulated for an advanced gas-cooled reactor, no form of iodine was found which could not be trapped by a combination of high efficiency filter and charcoal absorber, provided this device was suitably designed and operated. Nor have the early stages of an investigation of water reactor conditions produced any new, highly penetrating forms.

Iodine vapour is reactive and attaches itself so firmly to the impurities in some charcoals that it is well retained even when other vapours compete with it for space on the charcoal surface. Methyl iodide, however, is much less reactive; much less of it can be held on the same weight of charcoal and it can be more easily displaced by competing vapours. It is therefore essential to choose the charcoal more carefully and to test it against the mixture of gases which it is expected to decontaminate; a charcoal that is good, for instance, at holding iodine at high temperature is not necessarily good at trapping methyl iodide.

There has been in the past a shortage of well-documented experiments relevant to conditions postulated for reactor accidents and particularly of evidence as to the nature and trappability of the fission product compounds formed. For instance, there appears even now to be little known about release from steel-clad uranium-dioxide heated in steam to a temperature greater than the melting point of the can.

Nevertheless, as reflected especially in papers P/282 and P/285, there is a growing interest in such experiments and in the development of a number of interesting techniques of classification. I look forward to the day when reliable performance data are available to reactor designers.

Paper P/382

DISCUSSION

W. J. McCARTHY (United States of America): Has the analytical model for the slowing down of the centrifugal pump described in the paper been verified by experiment or test? More generally, what experimental work has been conducted in relation to this theoretical study?

G. N. POLETAEV (USSR): Experimental work has been carried out to a limited extent, but this is not covered by the paper. The experimental results agreed satisfactorily with theory.

Paper P/283 (presented by W. E. Nyer)

DISCUSSION

F. C. BOYD (Canada): In natural uranium reactors, such as those using heavy water, the excess reactivity is so small that the large excursions studied in this paper are *not* possible.

Do you now have a full explanation for the destructive pressure transient that occurred in the SPERT 1 D-core (Fig. 6) after the power burst?

W. E. NYER (United States of America): At the time of the development of the destructive pressure pulse, the fuel temperatures were such that the inner core of the plates was molten and this melted region had spread to the outer edge of the cladding. The tensile strength was negligible and the outside of the plates was above boiling temperature, so that the plate was virtually molten. In these conditions, it remains in its original configuration only until a disturbance causes a break-up of the liquid metal and disperses it into the water. A number of sources for such disturbance exist and any one of them could be responsible: surface tension, the thermal stresses on unmelted parts of the plate, or the mechanism vibrations in the system. We have observed in some experiments that plates occasionally come into contact with one another, trapping water and causing minor local steam explosions which could easily be the cause of initiating the break-up of plates in the molten state. This break-up and dispersion of droplets into intimate contact with the water is responsible for the pressure surge. We see no problem in understanding this aspect of the destructive pressure pulse. I refer you to report IDO-16884 by Houghtaling, Sola and Spano for additional details (issued by Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho, 1963).

Paper P/525 (presented by J. Hulovec)

DISCUSSION

W. E. NYER (United States of America): Are there any plans to make experimental comparisons with the results of the study or have any such experimental studies been made? Secondly, in relation to the brief reference to pipe rupture in the paper, what sizes of pipe rupture were considered in the analysis? Has a complete severance been considered?

J. HULOVEC (Czechoslovakia): The dynamic properties of the station will, of course, be checked experimentally. We have no reactor as yet, but the steam generators are already being run on the test stand.

We have considered the case of complete severance of one of the twelve outgoing pipes.

Fault detection and analyses

Paper P/436 (presented by A. Oyama)

DISCUSSION

J. M. YELLOWLEES (United Kingdom): The clad rupture temperature (T_a, Fig. 6 in the paper) must be associated with time on the basis of creep rupture. What criteria governed the selection of time?

Please explain the analytical assumptions used in calculating the clad temperature transient shown on Fig. 12. For instance, radiation heat transfer between the hotter and cooler fuel pins may be important when the spray distribution is poor.

A. OYAMA (Japan): This point was not covered in the oral presentation because of the time limitation. We did, of course, apply certain assumptions in calculating the transient of clad temperature but clad rupture is a very complicated phenomenon that will have to be investigated further.

S. LEVY (United States of America): Has any attempt been made to extrapolate this work to larger systems? Was consideration given to the transient pressures developed across the core? In your spray cooling tests, were you ever unable to cool any portion of the heated rods?

A. OYAMA (Japan): In answer to the first question, we have no plans at the moment for building this experimental apparatus on a larger scale. We would like first to do further fundamental study on the small-scale apparatus and perhaps two years hence may consider scaling it up. Dr. Uchida, the co-author of the paper, will be better able to answer the other question.

H. UCHIDA (Japan): On Mr. Levy's second question, the pressure transients in the four channels have to be determined by experiment, which presents difficulty from the standpoint of precision; in this connection we find his paper (P/283) of interest.

On the third point, with this particular spray cooling system, there is difficulty in getting effective cooling of the lower part of the heated fuel rods. We are in course of studying and developing a more efficient system for this reactor.

F. R. FARMER (Chairman): Dr. Lieberman, as spokesman for the United States safety research programme, would you like to add anything in answer to Dr. Levy's questions?

J. A. LIEBERMAN (United States of America): Pressure drops across the core have been looked at under the United States programme in connection with the analyses for LOFT (Loss of Fluid Tests). It appears that these pressure drops, following a loss-of-coolant accident, can develop considerable forces which may result in mechanical deformation or other adverse effects on the grid support structure, etc. Accordingly, it would seem important to give this subject perhaps somewhat more consideration than heretofore.

J. E. R. HOLMES (United Kingdom): This is in relation to the question of emergency spray cooling and is by way of a comment. For the SGHW light-water cooled reactor, we have considered it desirable to include an emergency cooling system to remove decay heat. We heave found that it is necessary to ensure efficient cooling over the full pressure range. With highly rated systems, our experiments have shown that the spray injection arrangements have to be carefully designed to get efficient and uniform cooling. I think this is consistent with the experience which you have described.

Paper P/517 (presented by G. Stiennon)

DISCUSSION

S. LEVY (United States of America): We would be interested in knowing whether any failures of the type considered in the paper have been experienced. Have you attempted to predict the temperature associated with the flow reversal? If not, do you think that the present results can be applied to other reactors?

G. STIENNON (Belgium): We have not yet had an accident similar to those simulated. We have had pressure losses, but up till now these have occurred on start-up when power was zero. We regard our calculations as being on the safe side and are also of the opinion that they can be applied to systems other than the BR2 reactor. We are applying them to different configurations in that reactor.

D. STOIAN (Belgium): As the person responsible for the thermal calculations, I can confirm that opinion, since conventional formulae only were used in the study of the various flow systems. Detailed information on this subject is given in the published report GEX/D/5.

D. F. CARROLL (South Africa): From the text of this paper I gather that actual measurements were made of the heat transfer coefficient within a fuel element, which were then compared with calculated values, as can be inferred from Fig. 5. If this is correct, what equation was used for calculating the heat-transfer coefficient used in Fig. 5 and what is the approximate correlation between the calculated and the measured values?

G. STIENNON (Belgium): No special measurement of the heat-transfer coefficient was made; using the conventional formulae, we derived it from the general measurements made.

GENERAL DISCUSSION

Fission product release and behaviour

F. C. BOYD (Canada): This is a relatively broad question, addressed to the authors of papers P/282, P/285 and P/186. Is the knowledge of fission-product transport and removal phenomena now sufficient to be applied to containment and general safety design? Would the authors comment on whether or not these phenomena can be depended upon to prevent the release of fission products to the environment?

J. A. LIEBERMAN (United States of America): The answer, very shortly, is "No", but Mr. Cottrell, who is more familiar with this aspect of the programme, can perhaps comment further.

W. B. COTTRELL (United States of America): The general answer to this question is in the negative, although there are many specific situations in which the behaviour of accidentally released fission products are known. The rather extensive programme in the United States has to date identified the amounts of the various isotopes which may be released under a wide variety of accident conditions. However, the transport of these fission products in containment systems is also a function of the physical and chemical forms of the released activity as well as of the characteristics of the containment system and the accident environment. Studies relating to the characterization of fission products and their behaviour under such conditions are a part of the Nuclear Safety programme at Oak Ridge National Laboratory, where various aspects of the problem are examined in a number of facilities, including the Nuclear Safety Pilot Plant. Such studies will be extended by the Containment Systems Experiment at Hanford and the Loss-of-Flow Tests at Phillips Petroleum (NRTS). These programmes and results to date are discussed at grater length in paper P/285.

R. D. COLLINS (United Kingdom): We do not know how to design systems which will adequately prevent all fission-product releases, but we do believe that in at least one system, the advanced gas-cooled reactor, we have identified the principal difficulties, have shown the conditions which must be met in design and operations, and have demonstrated that effective gas treatment can be relied on.

F. R. FARMER (Chairman): Dr. Lieberman, would you wish to add from the United States side at this point? I was a little surprised at your unqualified negative in the first place. I believe that most questions on safety would call for a negative answer in the absolute sense but that there exists a growing degree of confidence in a number of fields.

J. A. LIEBERMAN (United States of America): The negative was not meant to be categorical. I do not think that there is complete ignorance in this sphere. The efforts being made in a number of countries to characterize the nature and the behaviour of the fission products which could be released from reactor fuel provide a significant body of information, on the basis of which the consequences of a possible accident in a given reactor system may be assessed with greater confidence than heretofore and the results applied to design of a containment or air-cleaning system.

F. C. BOYD (Canada): I would not like to leave any wrong impression about the Canadian attitude to or knowledge of containment systems. We feel that, for a specific reactor system, it is possible to analyse expected releases of fission products sufficiently to enable special containment features to be designed that will be adequate for that particular system.

W. B. COTTRELL (United States of America): It is indicated in paper P/186 that the proportion of fission products released from the fuel was considerably greater than that found by Parker * in 1962 for releases in an inert gas. The release of fission products is a function of time, temperature and radiation exposure. Parker's more recent experimental work on fission product release (some of which is reported in paper P/285) has resulted in higher releases than were reported in the 1962 paper and are consistent with the 30–50% iodine release reported in paper P/186 (cf. in particular Fig. 1 in paper P/285).

On the whole, paper P/186 represents an excellent piece of work. I would like to know more of the unique physical properties of the type 207B charcoal which makes it so adsorbent for methyl iodide.

R. D. COLLINS (United Kingdom): We are not able to specify the characteristic of the coal-base charcoal which makes it able to decompose methyl iodide effectively.

W. J. MCCARTHY (United States of America): Has any work specifically aimed at the estimation of releases of fission products from sodium or sodium/potassium-alloy cooled systems been undertaken in the United Kingdom?

R. D. COLLINS (United Kingdom): At present, we have no major experiments on the retention of fission products in sodium systems.

Graphite oxidation and reactor satety

W. J. MCCARTHY (United States of America): It was indicated in paper P/185 that reactivity to oxygen of the graphite in the reactor moderator increased with neutron irradiation. Has any estimate been made of the dependence of the increase in reactivity on total

* Parker, G. W., *et al.*, *Parametric Studies of Fission Product Release from UO_2 Fuels*, Third Conference on Nuclear Reactor Chemistry, USAEC report, TID-7641 (1962).

dose rate, i.e., for a dose twice as great would the reactivity be twice as great or is this a saturable effect?

J. R. BEATTIE (United Kingdom): In paper P/185 the effect of neutron dose is given as about 0.4 μg/gh at 1 000 MWd/Adj t, and this is an approximately linear effect so far as is known at present. The irradiation effect on reactivity is believed to be a combination of two effects, one deriving from the neutron dose and the other from radiolytic corrosion by CO_2. As stated in the paper, the latter is believed to have as its maximum effect a fourfold increase in reactivity. Further work is going on to elucidate and clarify both effects and it is our intention to correlate this work with measurements on graphite from reactors as exposure proceeds.

Burst cartridge detection systems

J. A. LIEBERMAN (United States of America): The discussion of the detecting gear given in paper P/142 is extremely interesting. We could appreciate some indication of the percentage contribution of this system to total plant cost. What level of maintenance force is required to keep the system in operation? And lastly, what is the frequency of can failure?

B. HOLMES (United Kingdom): The question seems to be based on the assumption that a large quantity of equipment is needed. However, the Sizewell station does not call for a significant increase in equipment compared with earlier stations. The capital cost of the BCD gear is of the order of between 1% and 2% of station cost. As regards maintenance, the major problem relates to the electronic control and instrumentation equipment which, in fact, constitutes but a small part of this type of equipment on the station. Accordingly it is not possible to distinguish at this stage between anticipated maintenance personnel requirements for BCD equipment alone. Again the mechanical equipment is essentially that associated with charge chutes and its maintenance will therefore be part of general reactor equipment maintenance.

G. W. GREENWOOD (United Kingdom): The number of fuel elements in which failures have been confirmed from the United Kingdom CEGB reactors at Berkeley and Bradwell has been about 1 in 10 000. In the Magnox cans, there is so far no indication of fuel element failure due to cavitation, which was the cause of failure of some early fuel elements in the Calder Hall reactors. Some fuel elements have been discharged which have been shown to be sound on post-irradiation examination, suggesting that there is little to be gained by increasing the absolute sensitivity of the BCD gear.

J. WEILL (France): The sensitivities for burst cartridge detection given in Table 1 of paper P/142, particularly in the case of the Sizewell power station, seem surprising. Were they obtained by calculation or experiment? If the latter, how does it come about that superficial contamination of the fuel elements doesn't give out a higher background count? Has it been possible to reduce this contamination as compared with the results hitherto obtained, which were of the order of 0.01 $\mu g/cm^2$?

B. HOLMES (United Kingdom): The figure suggested by the questioner for can contamination was used in the calculation of sensitivity quoted in the paper. This figure was originally based on an experimental figure, the accuracy of which has subsequently been shown to be doubtful. Recent experiments have suggested a higher figure. However, it has also been found that this error was, in the case of the Calder Hall and Chapelcross reactors, offset by a discrepancy in the correlation of actual system performance with predicted performance. For this reason the background count is still based on the original calculations.

Compte rendu de la séance 3.9

Recherche en matière de sûreté nucléaire

Président: F. R. Farmer (Royaume-Uni)

Etudes sur la cinétique et les états transitoires dans les réacteurs

Mémoire P/282 (présenté par J. A. Lieberman)

DISCUSSION *

B. G. DUBOVSKY (URSS): Dans les projets de réacteurs rapides, tenez-vous compte de la possibilité d'un effet de réactivité positif (dû à un déplacement accidentel du sodium réfrigérant et au durcissement correspondant du spectre de neutrons) qui se produirait en même temps qu'un incident au système de sécurité? Quelle protection pratique prévoyez-vous dans le projet?

J. A. LIEBERMAN (Etats-Unis d'Amérique): Je pense que M. McCarthy pourrait répondre à cette question.

* Voir aussi la discussion générale dans la présente séance.

W. J. McCarthy (Etats-Unis d'Amérique): La possibilité d'un coefficient Doppler positif dû à la perte de sodium a été signalée pour la première fois par Okrent et Dickerman. Cette possibilité n'a pas été spécifiquement envisagée dans les projets de réacteurs de 1 000 MW(e), bien que l'on ait prévu les habituelles précautions très strictes contre les pertes de sodium, générales ou locales. Dans tout projet plus poussé il faudrait envisager en détail cette possibilité.

B. G. Dubovsky (URSS): Compte tenu de l'expérience que vous avez acquise, peut-on évaluer l'utilité des grandes enceintes de protection pour les réacteurs, en vue d'éviter la diffusion de la contamination radioactive en cas d'accident?

J. A. Lieberman (Etats-Unis d'Amérique): On a construit et utilisé avec succès aux Etats-Unis un certain nombre d'enceintes de confinement pour des réacteurs de puissance à eau. On a fait des essais détaillés concernant les taux de fuite, et des taux de fuite journalière de 0,1% du volume contenu, ou inférieurs, ont été obtenus. Vous pouvez vous référer à un manuel sur le confinement qui va être publié par Cottrell *et al.* du Laboratoire national d'Oak Ridge, et qui pourrait vous intéresser à propos de ce problème.

W. Häfele (République fédérale d'Allemagne): L'air rejeté à l'atmosphère hors de la plupart des enceintes de confinement passe à travers des batteries de filtres. Normalement, ces filtres ne peuvent supporter qu'une faible variation de pression. Travaille-t-on aux Etats-Unis à la réalisation de filtres résistant à la pression à haute température? Quelles sont les limites de pression et de température des meilleurs filtres disponibles? Quels sont les facteurs de filtration correspondants?

J. A. Lieberman (Etats-Unis d'Amérique): C'est une question importante. Il faut savoir comment un système d'épuration de l'air se comportera en cas d'accident, c'est-à-dire avec des pressions et des températures anormales. Les filtres à air absolus (du type en papier plissé) d'utilisation courante ne peuvent supporter que des pressions de quelques centimètres d'eau et des températures de quelques centaines de degrés Fahrenheit. L'un des objectifs importants de l'étude décrite dans notre mémoire est d'obtenir des données physiques sur le comportement d'un tel système dans des conditions aussi voisines que possible de celles qui suivraient un accident important.

Les dispositifs d'épuration d'air généralement utilisés ont des efficacités supérieures à 99,8% pour les particules de 0,3 micron de diamètre, et d'environ 99,95% pour l'iode, dans les conditions de fonctionnement normales.

K. A. Mahmoud (République arabe unie): Je vois dans votre mémoire que votre budget pour les études de sûreté nucléaire est de 18 millions de dollars pour l'année 1964. Quel pourcentage ceci représente-t-il du budget annuel consacré aux applications pacifiques de l'énergie atomique aux Etats-Unis? Je voudrais aussi connaître le pourcentage du personnel affecté aux études de sécurité nucléaire.

J. A. Lieberman (Etats-Unis d'Amérique): Bien que je ne dispose pas de chiffres précis ou de pourcentages, je peux vous dire que le pourcentage du budget total affecté à la recherche en matière de sûreté nucléaire est très petit, sans doute de l'ordre de 4%. Cependant, ceci ne comprend pas les études de sûreté faites en liaison avec des projets spécifiques de réacteurs, ni les études consacrées à la santé et à la surveillance de l'environnement.

Le nombre de personnes participant au programme décrit peut être estimé à 400–500.

R. D. Collins (Royaume-Uni) *: Dans la section sur le comportement des produits de fission du mémoire P/282, il est dit que dans certaines conditions l'iode tend à réagir avec des impuretés pour former des composés qui ne sont pas efficacement arrêtés par les systèmes usuels d'épuration de l'air. Il est exact qu'il peut se former des composés plus difficiles à arrêter que la vapeur d'iode, et, comme l'indique le mémoire P/186, nous avons identifié certains d'entre eux, dont l'iodure de méthyle qui est le plus important. Mais des expériences en nombres considérables, choisies de façon à inclure toutes les conditions correspondant à des accidents envisagés pour un réacteur à gaz de type avancé, ont montré qu'il n'y avait aucune forme d'iode qui ne puisse pas être arrêtée par une combinaison de filtre à haute efficacité et d'absorbeur à charbon actif, à condition que le dispositif soit bien conçu et fonctionne correctement. De plus, les premières études sur les conditions existant dans les réacteurs à eau n'ont mis en évidence aucune forme nouvelle très pénétrante.

La vapeur d'iode entre facilement en réaction et se fixe si énergiquement sur les impuretés dans certains charbons actifs qu'elle reste fixée même quand d'autres vapeurs sont en compétition avec elle pour la surface du charbon actif. L'iodure de méthyle, par contre, est beaucoup moins réactif; on ne peut en fixer qu'une quantité beaucoup plus faible sur le même poids de charbon actif, et il peut être plus facilement déplacé par d'autres vapeurs. Il est donc essentiel de choisir le charbon actif avec plus de soin et de l'essayer avec le mélange de gaz qu'il peut avoir à décontaminer; un charbon actif qui peut, par exemple, retenir l'iode à haute température, n'est pas nécessairement bon pour arrêter l'iodure de méthyle.

On a manqué dans le passé d'expériences bien documentées correspondant aux conditions envisagées pour des accidents de réacteurs, et en particulier de données sur la nature et la facilité de fixation des composés de produits de fission formés. Par exemple, il semble que

* Ces observations n'ont pas été faites en séance.

l'on ne sait encore presque rien de ce que dégagerait de l'oxyde d'uranium gainé en acier et chauffé dans de la vapeur à une température supérieure au point de fusion de la gaine.

Cependant, comme le montrent particulièrement les mémoires P/282 et P/285, on accorde un intérêt de plus en plus grand à ces expériences et à la mise au point d'un certain nombre de techniques intéressantes de classification. Il faut attendre avec espoir le jour où les ingénieurs qui établissent les plans des réacteurs disposeront de données sûres sur les performances de ces systèmes.

Mémoire P/382

DISCUSSION

W. J. McCARTHY (Etats-Unis d'Amérique): Le modèle analytique pour le ralentissement de la pompe centrifuge décrit dans le mémoire a-t-il été vérifié par des expériences ou des essais? Plus généralement, quel travail expérimental avez-vous fait en liaison avec cette étude théorique?

G. N. POLETAEV (URSS): Nous avons fait une étude expérimentale limitée, mais elle n'est pas décrite dans le mémoire. Les résultats expérimentaux sont en bon accord avec la théorie.

Mémoire P/283 (présenté par W. E. Nyer)

DISCUSSION

F. C. BOYD (Canada): Dans les réacteurs à uranium naturel, tels que ceux qui utilisent l'eau lourde, l'excès de réactivité est si faible que les grandes excursions de puissance étudiées dans ce mémoire ne sont pas possibles.

Pouvez-vous maintenant expliquer complètement l'augmentation transitoire de pression destructrice qui s'est produite dans le cœur D de SPERT 1 (figure 6) après la saute de puissance?

W. E. NYER (Etats-Unis d'Amérique): Au moment où l'impulsion de pression destructrice s'est manifestée, les températures du combustible étaient telles que l'intérieur des plaques était fondu, et cette partie fondue s'est répandue sur l'extérieur du gainage. La résistance à la traction était négligeable et l'extérieur des plaques était à une température supérieure à la température d'ébullition, de sorte que la plaque était virtuellement fondue. Dans ces conditions, une plaque ne conserve sa configuration originale que jusqu'au moment où une perturbation provoque la séparation dans le métal liquide et le disperse dans l'eau. Il y a un certain nombre de causes possibles pour une telle perturbation, et chacune d'entre elles pourrait provoquer ce résultat: tension superficielle, efforts thermiques sur les parties non fondues de la plaque, ou vibrations mécaniques du sytème. Nous avons observé que dans quelques expériences les plaques sont parfois au contact l'une de l'autre en emprisonnant de l'eau et en provoquant de petites explosions locales de vapeur qui pourraient facilement être la cause du début de rupture des plaques déjà à l'état fondu. Cette rupture et la dispersion de gouttelettes en contact intime avec l'eau sont responsables de la poussée de pression. La compréhension de cet aspect de la poussée de pression destructrice ne pose pas de problème. Vous pouvez vous référer au rapport IDO-16884, par Houghtaling, Sola et Spano (publié par la Philips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho, 1963), pour obtenir des détails complémentaires.

Mémoire P/525 (présenté par J. Hulovec)

DISCUSSION

W. E. NYER (Etats-Unis d'Amérique): Prévoyez-vous des comparaisons expérimentales avec les résultats de l'étude ou avez-vous déjà fait de telles études expérimentales? Deuxièmement, à propos de la brève référence à une rupture de canalisation, dans votre mémoire, quelles tailles de ruptures de canalisations avez-vous envisagé dans cette analyse? Avez-vous envisagé une séparation complète?

J. HULOVEC (Tchécoslovaquie): Les propriétés dynamiques de la centrale seront naturellement vérifiées expérimentalement. Nous n'avons pas encore de réacteur, mais les générateurs de vapeurs sont déjà en cours d'essai.

Nous avons envisagé le cas de la section complète de l'une des douze canalisations de sortie.

Détection et analyse des incidents

Mémoire P/436 (présenté par A. Oyama)

DISCUSSION

J. M. YELLOWLEES (Royaume-Uni): La température de rupture de gaine (T_d, figure 6 du mémoire) doit être liée au temps sur la base de la rupture par fluage. En fonction de quels critères avez-vous choisi le temps?

Pouvez-vous expliquer les hypothèses analytiques utilisées pour le calcul de la température de gaine transitoire qui figure sur la figure 12? Par exemple, le transfert de chaleur par rayonnement entre les aiguilles de combustible chaudes et froides peut devenir important si la distribution de vaporisation est mauvaise.

A. OYAMA (Japon): Le temps limité imparti à la présentation orale n'a pas permis de traiter ce point. Nous avons, bien entendu, fait certaines hypothèses pour calculer la température de gaine transitoire, mais la rupture de gaine est un phénomène compliqué qui nécessitera d'autres études.

S. LEVY (Etats-Unis d'Amérique): Avez-vous essayé d'extrapoler ce travail à des systèmes plus importants? Avez-vous envisagé des pressions transitoires se développant à travers le cœur? Dans vos essais de refroidissement par vaporisation, vous est-il arrivé de ne pas pouvoir refroidir certaines parties des barres chauffées?

A. OYAMA (Japon): En réponse à votre première question, je dirai que nous n'envisageons pas pour le moment de construire cette installation expérimentale à plus grande échelle. Nous voudrions d'abord poursuivre l'étude fondamentale sur l'installation de petite taille, et nous pourrons peut-être envisager dans deux ans une installation plus grande. Le Dr Uchida, co-auteur du mémoire, pourra mieux répondre aux autres questions.

H. UCHIDA (Japon): En réponse à la seconde question de M. Levy, je dirai que les transitoires de pression dans les quatre canaux doivent être déterminés expérimentalement, ce qu'il est difficile de faire avec précision; à ce sujet, nous sommes très intéressés par son mémoire (P/283).

En ce qui concerne la troisième question, il est difficile, avec ce système particulier de refroidissement par vaporisation, de refroidir efficacement la partie inférieure des éléments combustibles chauffés. Nous sommes en train d'étudier et de mettre au point un système plus efficace pour ce réacteur.

F. R. FARMER (Président): Le Dr Lieberman, en tant que porte-parole pour le programme des études de sécurité aux Etats-Unis, voudrait-il faire quelques observations, en rapport avec des questions du Dr Levy?

J. A. LIEBERMAN (Etats-Unis d'Amérique): Les chutes de pression dans le cœur ont été étudiées au titre du programme des Etats-Unis en liaison avec les analyses pour le programme LOFT (*Loss of Fluid Tests*, Essais de perte de fluide). Il semble que ces chutes de pression, à la suite d'un accident avec perte de réfrigérant, peuvent produire des forces considérables qui peuvent entraîner des déformations mécaniques ou d'autres effets néfastes sur la grille de support, etc. C'est pourquoi il semble qu'il serait utile de consacrer à cette étude plus d'attention qu'on ne lui en a accordé jusqu'à présent.

J. E. R. HOLMES (Royaume-Uni): Je voudrais faire une remarque à propos du refroidissement de secours par vaporisation. Pour le réacteur refroidi à l'eau légère SGHW, nous avons jugé souhaitable d'avoir un système de refroidissement de secours pour éliminer la chaleur de décroissance. Nous avons trouvé qu'il était nécessaire d'assurer un refroidissement efficace dans tout le domaine des pressions. Avec des systèmes de puissance spécifique élevée, nos expériences ont montré que les dispositifs d'injection et de vaporisation doivent être conçus avec soin pour obtenir un refroidissement efficace et uniforme. Ceci me semble tout à fait conforme à ce que vous avez décrit.

Mémoire P/517 (présenté par G. Stiennon)

DISCUSSION

S. LEVY (Etats-Unis d'Amérique): Il serait intéressant de savoir si des incidents du type envisagé dans ce mémoire se sont produits. Avez-vous essayé de prédire la température associée à l'inversion d'écoulement? Sinon, pensez-vous que vos résultats peuvent être appliqués à d'autres réacteurs?

G. STIENNON (Belgique): Nous n'avons pas encore eu d'accident analogue à ceux qui sont simulés. Nous avons eu des chutes de pression, mais jusqu'à présent elles se sont produites au moment du démarrage quand la puissance était nulle. Nous pensons que nos calculs présentent une bonne marge de sécurité, et pensons aussi qu'ils pourraient s'appliquer à des systèmes autres que le réacteur BR-2. Nous les appliquons à différentes configurations dans ce réacteur.

D. STOIAN (Belgique): En tant que responsable des calculs thermiques, je peux confirmer cette opinion, car nous n'avons utilisé que des formules classiques dans l'étude des divers systèmes d'écoulement. Le rapport publié GEX/D/5 donne des renseignements détaillés sur ce sujet.

D. F. CARROLL (Afrique du Sud): D'après le texte du mémoire il semble que vous avez effectivement mesuré les coefficients de transfert thermique à l'intérieur d'un élément de combustible, et comparé les résultats aux valeurs calculées, comme l'indique la figure 5. Si ceci est exact, quelle équation avez-vous utilisée pour calculer le coefficient de transfert thermique utilisé dans la figure 5, et quelle est approximativement la corrélation entre les valeurs calculées et mesurées?

G. STIENNON (Belgique): Nous n'avons pas spécialement mesuré le coefficient de transfert thermique; à l'aide des formules classiques, nous l'avons déduit des mesures générales qui ont été faites.

DISCUSSION GÉNÉRALE

Dégagement et comportement des produits de fission

F. C. BOYD (Canada): Je voudrais poser une question assez générale, qui s'adresse aux auteurs des

mémoires P/282, P/285 et P/186. A-t-on maintenant des connaissances suffisantes des phénomènes de transport et d'élimination des produits de fission pour pouvoir les appliquer à l'étude du confinement et de la sûreté en général? Les auteurs pourraient-ils indiquer si on peut s'appuyer sur ces phénomènes pour éviter la dispersion des produits de fission dans le milieu environnant?

J. A. LIEBERMAN (Etats-Unis d'Amérique): La réponse, très brièvement, est « Non », mais M. Cottrell, qui est plus familier avec cet aspect du programme, pourrait peut-être préciser cette réponse.

W. B. COTTRELL (Etats-Unis d'Amérique): La réponse générale à cette question est négative, bien qu'il y ait un grand nombre de situations spécifiques pour lesquelles on connaît le comportement des produits de fission accidentellement dégagés. Le programme assez complet des Etats-Unis a permis d'identifier à ce jour les quantités des divers isotopes qui peuvent être dégagées dans une gamme importante de conditions exceptionnelles. Cependant, le transport de ces produits de fission dans les systèmes de confinement dépend aussi des formes physiques et chimiques de l'activité dégagée, ainsi que des caractéristiques du système de confinement et du milieu environnant. Les études liées à la caractérisation des produits de fission et de leur comportement dans de telles conditions font partie du programme de sûreté nucléaire du Laboratoire national d'Oak Ridge, où divers aspects de ce problème sont étudiés dans un certain nombre d'installations, y compris l'usine pilote de sûreté nucléaire. Des études seront complétées par l'Expérience de systèmes de confinement à Hanford, et les Essais de perte d'écoulement par Philips Petroleum (NRTS). Ces programmes et les résultats déjà obtenus sont décrits plus en détail dans le mémoire P/285.

R. D. COLLINS (Royaume-Uni): Nous ne savons pas établir les plans de systèmes capables d'empêcher efficacement tous les dégagements de produits de fission, mais nous pensons que dans un cas au moins, celui du réacteur à gaz de type avancé, nous avons identifié les principales difficultés, montré les conditions auxquelles doivent satisfaire les plans et le fonctionnement, et démontré que l'on peut se fier à un traitement efficace du gaz.

F. R. FARMER (Président): Le Dr Lieberman voudrait-il faire quelques observations représentant le point de vue des Etats-Unis? J'ai été un peu surpris de sa première réponse entièrement négative. Je pense que la plupart des questions sur la sûreté appellent une réponse négative au sens absolu, mais dans un grand nombre de domaines le degré de confiance va en augmentant.

J. A. LIEBERMAN (Etats-Unis d'Amérique): Ma réponse négative ne devrait pas être considérée comme catégorique. Je ne pense pas que nous ignorons tout dans ce domaine. Les efforts en cours dans un certain nombre de pays en vue de caractériser la nature et le comportement des produits de fission qui pourraient se dégager du combustible d'un réacteur ont donné une masse importante de renseignements, qui permettent d'évaluer avec plus de confiance qu'auparavant les conséquences d'un accident éventuel dans un système donné de réacteur, et les résultats de ces études permettent de concevoir des systèmes de confinement ou d'épuration de l'air.

F. C. BOYD (Canada): Je ne voudrais pas donner une fausse impression de l'attitude canadienne envers les systèmes de confinement ou de la connaissance que nous en avons. Nous pensons que pour un système particulier de réacteur, il est possible d'analyser suffisamment bien les dégagements prévus de produits de fission pour permettre de concevoir des dispositifs particuliers de confinement qui seront tout à fait suffisants pour ce système particulier.

W. B. COTTRELL (Etats-Unis d'Amérique): Il est indiqué dans le mémoire P/186 que la proportion de produits de fission dégagée du combustible a été beaucoup plus importante que celle qui avait été observée par Parker * en 1962 pour le dégagement sous gaz inerte. Le dégagement des produits de fission est fonction du temps, de la température et de l'irradiation. Les résultats expérimentaux les plus récents de Parker sur le dégagement des produits de fission (dont une partie est publiée dans le mémoire P/285) ont abouti à des dégagements plus importants que ceux qui avaient été indiqués dans la publication de 1962, et qui sont compatibles avec le dégagement de 30–50% d'iode signalé dans le mémoire P/186 (voir en particulier la figure 1 du mémoire P/285).

Dans l'ensemble, le mémoire P/186 représente un excellent travail. Je voudrais avoir des détails sur les propriétés physiques exceptionnelles du charbon actif type 207B, qui le rendent si adsorbant pour l'iodure de méthyle.

R. D. COLLINS (Royaume-Uni): Nous ne sommes pas en mesure de spécifier les caractéristiques du charbon actif à base de charbon qui lui permettent de décomposer efficacement l'iodure de méthyle.

W. J. MCCARTHY (Etats-Unis d'Amérique): Avez-vous entrepris au Royaume-Uni des études spécifiquement destinées à estimer les dégagements de produits de fission dans des systèmes refroidis au sodium ou à l'alliage sodium-potassium?

R. D. COLLINS (Royaume-Uni): Pour le moment, nous n'avons pas fait d'expériences importantes sur la rétention des produits de fission dans les systèmes à sodium.

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* Parker, G. W. *et al.*, *Parametric Studies of Fission Product Release from UO_2 Fuels*, Third Conference on Nuclear Reactor Chemistry, USAEC report TID-7641 (1962).

Oxydation du graphite et sûreté des réacteurs

W. J. McCARTHY (Etats-Unis d'Amérique): Le mémoire P/185 indique que la réactivité du graphite du modérateur à l'égard de l'oxygène croît avec l'irradiation. Avez-vous essayé d'estimer la variation de la réactivité en fonction de la dose totale, autrement dit, est-ce que pour une dose deux fois plus grande, la réactivité serait deux fois plus grande, ou bien y-a-t-il un effet de saturation?

J. R. BEATTIE (Royaume-Uni): Le mémoire P/185 donne un effet de la dose de neutrons d'environ 0,4 μg/g h à 1 000 MWj par tonne adjacente, et l'effet est approximativement linéaire d'après nos connaissances actuelles. Il semble que l'effet de l'irradiation sur la réactivité est la combinaison de deux effets, l'un dû à la dose de neutrons et l'autre dû à la corrosion radiolytique par CO_2. Comme l'indique la communication, nous croyons que ce dernier effet, à son maximum, multiplie la réactivité par le facteur quatre. La suite des études permettra d'élucider et de préciser ces deux effets, et nous avons l'intention de comparer ces résultats à des mesures sur le graphite des réacteurs au fur et à mesure de l'irradiation.

Systèmes de détection de rupture de gaines

J. A. LIEBERMAN (Etats-Unis d'Amérique): La discussion de l'appareillage de détection donnée dans le mémoire P/142 est extrêmement intéressante. Nous serions heureux d'avoir une indication de la contribution en pourcentage de ce système au coût total de la centrale. Quel personnel d'entretien faut-il pour assurer le fonctionnement du système? Et enfin, quelle est la fréquence des ruptures de gaine?

B. HOLMES (Royaume-Uni): La question semble basée sur la supposition qu'il faut une quantité importante d'équipement. Mais il n'y aura pas pour la centrale de Sizewell d'accroissement notable de l'équipement par comparaison avec les centrales précédentes. L'investissement pour l'installation de détection de rupture de gaine représente entre 1 et 2% du prix de la centrale. En ce qui concerne l'entretien, le problème essentiel est celui de l'appareillage électronique et de l'instrumentation qui, en fait, ne représente qu'une faible partie de l'équipement de ce type dans la centrale. Il n'est donc pas possible de distinguer à l'heure actuelle les besoins en personnel d'entretien pour l'équipement de détection de rupture de gaine seul. Enfin, l'équipement mécanique est essentiellement celui qui est associé aux ensembles de chargement et son entretien fera donc partie de l'entretien de l'équipement général du réacteur.

G. W. GREENWOOD (Royaume-Uni): Le nombre d'éléments de combustible qui se sont effectivement révélés défectueux dans les réacteurs du CEGB du Royaume-Uni à Berkeley et Bradwell a été d'environ 1 pour 10 000. Dans les gaines Magnox, il n'y a encore eu aucune indication de rupture par cavitation, comme on l'avait observé sur certains des premiers éléments combustibles des réacteurs de Calder Hall. Certains éléments de combustible qui ont été déchargés se sont révélés en bon état au cours de l'examen après irradiation, ce qui suggère qu'il y a peu à gagner à augmenter la sensibilité absolue de l'appareillage de détection de rupture de gaine.

J. WEILL (France): Les sensibilités pour la détection de rupture de gaine indiquées dans le tableau 1 du mémoire P/146, en particulier dans le cas de la centrale de Sizewell, semblent surprenantes. Ont-elles été obtenues par le calcul ou par l'expérience? Si c'est par l'expérience, comment se fait-il que la contamination superficielle des éléments de combustible ne donne pas un bruit de fond plus élevé? A-t-il été possible de réduire cette contamination par rapport aux résultats obtenus jusqu'à présent, qui étaient de l'ordre de 0,01 $\mu g/cm^2$?

B. HOLMES (Royaume-Uni): Le chiffre que vous avez cité pour la contamination des gaines est celui qui a été utilisé dans le calcul de sensibilité cité dans le mémoire. Ce chiffre a été à l'origine basé sur une valeur expérimentale, dont l'exactitude a par la suite été jugée douteuse. Des expériences récentes ont suggéré une valeur plus forte. Cependant, on a également trouvé que cette erreur, dans le cas des réacteurs de Calder Hall et de Chapelcross, était compensée par un écart qui se manifeste dans la comparaison entre les performances réelles du système et les performances prédites. C'est pourquoi la valeur du bruit de fond est encore basée sur les calculs initiaux.

Протокол заседания 3.9

Исследование по ядерной безопасности

Председатель: Ф. Р. Фармер (Соединенное Королевство)

Исследование динамики и переходных процессов в реакторе

Доклад P/282 (представил Дж. А. Либерман)

ДИСКУССИЯ*

Б. Г. ДУБОВСКИЙ (СССР): Учитывается ли при проектировании реакторов на быстрых нейтронах возможность возникновения положительного коэффициента реактивности (из-за потери натриевого теплоносителя и последующего повышения средней энергии нейтронного спектра) и одновременного выхода из строя системы аварийной защиты? Какие инженерные решения используются для предотвращения такой маловероятной аварии?

Дж. А. ЛИБЕРМАН (США): На этот вопрос, по-видимому, может ответить г-н Маккарти.

У. Дж. МАККАРТИ (США): Возможность появления положительного доплеровского коэффициента реактивности в результате потери натрия была впервые показана Окрентом и Дикерманом. Такая возможность специально не рассматривалась при проектировании реакторов электрической мощностью 1000 *Мвт*. В конструкции учтены, конечно, обычные меры предосторожности против полной или частичной потери натрия. В любой серьезной конструкции такая возможность должна быть детально рассмотрена.

Б. Г. ДУБОВСКИЙ (СССР): На основе имеющегося у Вас опыта можно ли оценить целесообразность использования крупных противоаварийных оболочек реакторов, предназначенных для предотвращения распространения радиоактивных загрязнений при аварийной ситуации?

Дж. А. ЛИБЕРМАН (США): В США сооружены и успешно работают несколько противоаварийных оболочек для водяных реакторных установок. Проведены детальные исследования скорости утечки, достигнута суточная скорость утечки порядка 0,1% полного объема оболочки и даже ниже. Я могу порекомендовать Вам справочник, составленный Коттрелом и др., где Вы найдете ответ на интересующий вопрос.

* См. также общую дискуссию на этом заседании.

В. ХЕФЕЛЕ (ФРГ): Воздух, выбрасываемый из противоаварийной оболочки в атмосферу, проходит через ряд фильтров. Обычно эти фильтры работают при небольшом перепаде давления. Ведется ли в США разработка фильтров на большие перепады давления, способные работать при высоких температурах? Какие предельные температуры и давления выдерживают лучшие из существующих фильтров.?

Дж. А. ЛИБЕРМАН (США): Это очень важный вопрос. Прежде всего необходимо знать эффективность очистки воздуха в системе в случае аварии при давлении и температуре, отличающихся от обычных значений. Используемые в настоящее время абсолютные воздушные фильтры (типа гофрированных бумажных фильтров) могут выдерживать давление в несколько сантиметров водяного столба и температуру несколько сот градусов Фаренгейта. Одной из основных задач описанных в докладе исследований было получение технических данных о работе такой системы в условиях крупной аварии. В нормальных рабочих условиях эффективность очистки воздуха этими фильтрами составляет 99,8% для частиц диаметром 0,3 *мк* и около 99,95% для иода.

К. А. МАХМУД (ОАР): Как я понял из доклада, стоимость работ по ядерной безопасности в 1964 г. составила 18 млн. долларов. Какую часть эти затраты составляют от затрат на программу работ США по мирному использованию атомной энергии? Хотелось бы также узнать, какая часть персонала занята разработкой проблем ядерной безопасности.

Дж. А. ЛИБЕРМАН (США): Я не располагаю точными данными, но примерно можно сказать, что затраты на ядерную безопасность составляют порядка 4% от общих затрат. Однако сюда не включаются работы по безопасности, связанные с разработкой реакторов специального назначения, а также работы по технике безопасности и наблюдению за активными выпадениями. В работах по программе ядерной безопасности принимает участие приблизительно 400—500 человек.

* Это замечание на заседании не было представлено.

Р. Д. КОЛЛИНС (Соединенное Королевство): В разделе о поведении продуктов деления доклада P/282 указывается, что при определенных условиях иод имеет тенденцию к образованию соединений с некоторыми примесями, которые плохо удаляются современными системами воздушных фильтров. Такие соединения иода действительно образуются и они труднее улавливаются, чем пары иода (см. доклад P/186). Некоторые из этих соединений были нами обнаружены, наиболее важным из них является иодистый метил. Было выбрано значительное количество экспериментов, включающих основные из возможных аварийных случаев, присущих усовершенствованным реакторам с газовым охлаждением. Однако не было обнаружено соединений иода, которые не улавливались бы высокоэффективным фильтром в комбинации с древесным углем.

Такая система была разработана и испытана. Пары иода настолько прочно соединяются с поверхностными загрязнениями в древесном угле, что они не уносятся даже в том случае, когда пары других веществ начинают вытеснять их с поверхности древесного угля. Иодистый метил является химически менее активным, слабее удерживается тем же самым объемом древесного угля и легче замещается другими конкурирующими парами. Поэтому при проведении испытаний со смесью газов, которые подлежат дезактивации, нужно обратить серьезное внимание на выбор древесного угля. Например, древесный уголь, хорошо удерживающий иод при высокой температуре, может слабо улавливать иодистый метил. Недостаток всех прошлых экспериментов состоит в том, что они проводились в условиях, постулированных реакторных аварий и в особенности это относится к определению природы и способности удержания различных соединений продуктов деления. Например, даже сейчас очень мало известно о выделении продуктов деления из стальной оболочки двуокиси урана, нагретой в паре до температуры выше точки плавления. В связи с этим, как указывается в докладах P/282 и P/285, возрастает интерес к разработке надежных методик и проведению такого рода экспериментов. Я надеюсь, что в недалеком будущем конструкторы реакторов будут располагать надежными данными.

Доклад P/382 (представил Г. Н. Полетаев)

ДИСКУССИЯ

У. Дж. МАККАРТИ (США): Проводились ли какие-либо экспериментальные работы, относящиеся к этим теоретическим исследованиям?

Г. Н. ПОЛЕТАЕВ (СССР): Выполнено довольно ограниченное число экспериментов, которые не описаны в докладе. Экспериментальные результаты хорошо согласуются с расчетом.

Доклад P/283 (представил У. Э. Найер)

ДИСКУССИЯ

Ф. К. БОЙД (Канада): Тяжеловодные реакторы на естественном уране имеют настолько малую избыточную реактивность, что описанные в докладе большие отклонения, по моему мнению, невозможны. Как Вы объясните возникновение разрушающего импульса давления в активной зоне D реактора SPERT-1 (рис. 6) после импульса мощности?

У. Э. НАЙЕР (США): В момент возникновения разрушающего импульса давления температура топлива внутри активной зоны была настолько высокой, что пластинки расплавились, расплавленная область распространилась до внешней поверхности оболочки. Прочность на разрыв была незначительной, наружная поверхность пластин имела температуру выше точки кипения и пластины фактически были расплавлены. В этих условиях начальная конфигурация была разрушена, жидкий металл вытек и попал в воду. Причин разрушения было несколько, таковыми могли быть поверхностное натяжение, термические напряжения нерасплавленных участков пластин или механизм вибрации в системе. В некоторых опытах было обнаружено, что пластины случайно соприкасаются друг с другом и в результате кипения захваченной воды образуются небольшие локальные паровые взрывы, которые могут инициировать разрушение пластин в расплавленном состоянии. Такое разрушение и диспергирование капелек в воду является причиной возникновения импульса давления. Причины возникновения импульса давления нами установлены. Более детально этот вопрос излагается в отчете IDO-16884, составленном Хаугтелингом, Соула и Спэном (издан отделением «Атомик энерджи» фирмы («Филлипс петролеум», Айдахо Фоллз, Айдахо, 1963).

Доклад P/525 (представил Я. Гуловец)

ДИСКУССИЯ

У. Э. НАЙЕР (США): Имеются ли какие-либо установки, на которых будет проводиться экспериментальная проверка полученных результатов исследований? Какие масштабы указанных в докладе разрушений трубопроводов рассматривались в процессе анализа?

Я. ГЕЛОВЕЦ (Чехословакия): Динамические характеристики атомной электростанции, естественно, будут проверяться эксперимен-

тально. У нас еще нет реактора, но парогенератор уже проходит стендовые испытания. Что касается вопроса относительно разрыва трубопровода, то нами рассматривался случай полного выхода из строя одного из двенадцати выходных трубопроводов.

Обнаружение аварий и установление их причин

Доклад P/436 (представил А. Ояма)

ДИСКУССИЯ

Дж. М. ЙЕЛЛОУЛЕС (Соединенное Королевство): Температура разрыва оболочки (рис. 5 доклада) должна зависеть от времени на основе разрушения вследствие ползучести. Какие критерии играли основную роль при выборе времени? Объясните, пожалуйста, какие аналитические предпосылки использовались при расчете переходного температурного режима оболочки (рис. 12 доклада). Например, тепловое излучение при переходном процессе между более горячими и менее горячими стержнями может играть существенную роль в случае недостаточного разбрызгивания водяных капель.

А. ОЯМА (Япония): Это обстоятельство из-за недостатка времени не было учтено. Конечно определенные предпосылки использовались в расчетах переходного температурного режима оболочки. Разрушение оболочки очень сложное явление и в дальнейшем оно будет исследоваться более тщательно.

С. ЛЕВИ (США): Была ли сделана попытка экстраполировать результаты этой работы на более крупные системы? Рассматривался ли вопрос о давлении переходного режима в активной зоне? В Ваших испытаниях по охлаждению путем разбрызгивания воды можно ли было охлаждать какую-либо часть нагретых стержней?

А. ОЯМА (Япония): Отвечая на первый вопрос, я хочу сообщить, что мы не планировали создание крупных экспериментальных установок. Будут прежде всего проводиться фундаментальные исследования на небольших установках, на эти исследования, вероятно, потребуется два года. На следующий вопрос может дать более исчерпывающий ответ соавтор доклада доктор Ушида.

Х. УШИДА (Япония): Давление переходного процесса определялось экспериментально в четырех каналах. Эти измерения представляют определенные трудности с точки зрения точности результатов и в связи с этим мы сочли целесообразным включить этот вопрос в доклад P/283. По поводу частичного охлаждения системы можно сказать следующее: довольно трудно достигнуть эффективного охлаждения нижней части нагретых топливных стержней. Мы разрабатываем и испытываем более эффективную систему охлаждения для этого реактора.

Ф. Р. ФАРМЕР (председатель): Доктор Либерман, Вы как представитель США по вопросам исследования безопасности, не могли бы дать ответ на вопросы доктора Леви?

Дж. А. ЛИБЕРМАН (США): Перепады давления в активной зоне были измерены в соответствии с программой США по испытаниям с потерей теплоносителя. Оказалось, что перепад давления, возникающий после потери теплоносителя, может создавать значительные напряжения, которые могут привести к механической деформации или другим неблагоприятным эффектам в конструкции опорной решетки. Этот вопрос, вероятно, заслуживает более внимательного рассмотрения, чем это делалось до сих пор.

Дж. Э. Р. ХОЛМС (Соединенное Королевство): Несколько слов по вопросу аварийной системы охлаждения разбрызгиванием. Мы считали целесообразным для охлаждаемого водой реактора предусмотреть эффективную аварийную систему охлаждения, работающую без снижения давления для отвода остаточного тепла, образующегося в результате радиоактивного распада. Наши эксперименты показали, что сконструированная система инжекции брызг обеспечивает эффективное и равномерное охлаждение. Мне кажется, что эти данные согласуются с экспериментальными результатами, описанными в Вашем докладе.

Доклад P/517 (представил Г. Стайеннон)

ДИСКУССИЯ

С. ЛЕВИ (США): Какие неполадки были обнаружены при испытании описанного в докладе типа реактора? Была ли сделана попытка предсказать температуру, связанную с изменением потока? Если нет, то думаете ли Вы, что полученные Вами результаты могут быть использованы для других реакторов?

Г. СТАЙЕННОН (Бельгия): У нас подобного случая не было. Был случай, связанный с потерей давления, но это произошло во время пуска реактора, когда его мощность была равна нулю. Наши расчеты применительно к безопасности реактора мы считаем возможным использовать не только для реактора BR-2, но и для других систем. В частности, эти расчеты

мы используем для различных конфигураций такого реактора.

Д. СТОЯН (Бельгия): Как лицо, ответственное за тепловые расчеты, я могу подтвердить, что использовалась обычная формула только в изучении различных систем потока. Более детальная информация по этому вопросу опубликована в докладе GEX/D/5.

Д. КЭРРОЛЛ (ЮАР): На основании обсуждаемого доклада я сделал вывод, что коэффициент теплопередачи в тепловыделяющем элементе был измерен и сравнен с расчетными данными, как это можно понять из рис. 5. Если это правильно, то какое уравнение было использовано для расчета коэффициента теплопередачи (рис. 5) и каково расхождение между расчетными и измеренными результатами.

Г. СТАЙЕННОН (Бельгия): Специальных измерений коэффициента теплопередачи не производилось. В расчетах использовалось уравнение, выведенное на основе обычной формулы и известных экспериментов.

ОБЩАЯ ДИСКУССИЯ

Выделение и распространение продуктов деления

Ф. К. БОЙД (Канада): Я хочу задать вопрос авторам докладов P/282, P/285 и P/186. Достаточно ли хорошо изучен процесс распространения и удаления продуктов деления, чтобы можно было предусмотреть при разработке проектов соответствующие меры по безопасности установок? Могут ли авторы сказать, что этот процесс изучен настолько, что имеется возможность предотвратить распространение выделившихся продуктов деления в окружающую среду?

Дж. А. ЛИБЕРМАН (США): Я могу ответить на этот вопрос коротко «нет», но доктор Коттрелл, который более тесно связан с этими аспектами программы, может ответить подробнее.

У. Б. КОТТРЕЛЛ (США): В общем ответ на этот вопрос можно дать отрицательный, хотя имеется значительное количество специальных ситуаций, в которых поведение выделившихся в результате аварии продуктов деления известно. Целью ранней программы США было идентифицировать количества различных изотопов, которые могут выделяться при разных аварийных условиях. Однако распространение продуктов деления в противоаварийных системах в зависимости от физических и химических форм выделившейся активности является не менее важным, чем характеристики противоаварийной системы и загрязнение окружающего пространства. Изучение характеристик продуктов деления и их поведения в аварийных условиях составляет часть программы по ядерной безопасности Окриджской национальной лаборатории, где различные аспекты этой проблемы исследуются на экспериментальных установках, в том числе на опытной установке по исследованию ядерной безопасности. Подробные исследования будут проводиться на опытной противоаварийной системе в Ханфорде и фирмой «Филлипс петролеум» на Национальной станции по испытанию реакторов в Айдахо. Программа этих исследований и полученные результаты обсуждаются в докладе P/285.

Р. Д. КОЛЛИНС (Соединенное Королевство): Нам не известно, как можно спроектировать системы, обеспечивающие полное удержание выделившихся продуктов деления. Однако, как нам кажется, (по крайней мере для одной системы (усовершенствованный реактор с газовым охлаждением, принципиальные проблемы которого нами достаточно изучены) определены условия, которые могут встретиться при проектировании и эксплуатации, и продемонстрирована эффективная и надежная очистка газа.

Ф. Р. ФАРМЕР (председатель): Доктор Либерман, не желаете ли Вы что-либо добавить к вышесказанному? Я был несколько удивлен Вашим необоснованным отрицательным ответом в начале дискуссии. Я согласен, что на большинство подобных вопросов по безопасности можно дать отрицательный ответ, но степень нашей уверенности возрастает в ряде областей.

Дж. А. ЛИБЕРМАН (США): Мой ответ не означает категорическое отрицание. Я не думаю, что эта область остается совершенно неизвестной. Проведенные в некоторых странах исследования характеристик и поведения продуктов деления, которые могут выделяться из реакторного топлива, дали большое количество информации. Эти данные позволяют с большей, чем прежде, достоверностью оценить возможные аварийные ситуации для данных реакторных систем и использовать их при проектировании противоаварийных оболочек и систем очистки газа.

Ф. К. БОЙД (Канада): Я хотел бы сказать несколько слов о позиции Канады в вопросе противоаварийных систем. Для специальных реакторных систем, по нашему мнению, можно рассчитать ожидаемое выделение продуктов деления с точностью, достаточной для проектирования противоаварийной защиты, удовлетворяющей требованиям такой специфической системы.

У. Б. КОТТРЕЛЛ (США): В докладе P/186 указывается, что количество продуктов деления, выделившееся из топлива, было значитель-

но больше по сравнению с данными Паркера * по выделению продуктов в инертный газ. Выделение продуктов деления зависит от времени, температуры и длительности облучения. Результаты последних опытов Паркера по выделению продуктов деления (некоторые из них приведены в докладе P/285) показали количество выделившихся продуктов большее, чем было опубликовано в 1962 г., что составляет 30—50% от количеств, приведенных в докладе P/186 (в частности, сравните с рис. 1 доклада P/285). В целом доклад P/186 представляет исследования, выполненные на высоком уровне. Хотелось бы более подробно узнать замечательные физические свойства древесного угля марки 207 B, который так хорошо поглощает иодистый металл.

Р. Д. КОЛЛИНС (Соединенное Королевство): Я не могу точно указать характеристики древесного угля, который эффективно разлагает иодистый метил.

У. Д. МАККАРТИ (США): Проводится ли в Великобритании какая-либо работа по оценке выделения продуктов деления из натрия или сплава натрий — калий, используемых для охлаждения систем?

Р. Д. КОЛЛИНС (Соединенное Королевство): В настоящее время у нас не проводятся крупные эксперименты по удержанию продуктов деления из натриевых систем.

Окисление графита и безопасность реактора

У. Дж. МАККАРТИ (США): В докладе P/185 указывается, что с ростом дозы нейтронного облучения реакционная способность графита замедлителя к кислороду возрастает. Имеются ли какие-либо оценки зависимости роста реакционной способности от полной дозы облучения, то есть с увеличением дозы в два раза реакционная способность удвоится или имеет место эффект подавления реакционной способности?

Дж. Р БИТТИ (Соединенное Королевство): В докладе P/185 указано, что эффект нейтронной дозы составляет 0,4 *мкг/gh* при выгорании 1000 *Мвт · сутки/т*. По современным представлениям эта зависимость приблизительно линейная. Предполагают, что влияние облучения на реакционную способность представляет собой комбинацию двух эффектов, один из которых определяется нейтронной дозой, а другой — радиолитической коррозией за счет углекислого газа. Как указано в докладе, второй эффект, как полагают, приводит к четырехкратному увеличению реакционной способности. Ведется работа по определению механизма этих двух эффектов и установлению соответствия с измерениями в графитовом замедлителе реактора.

Детектирующее устройство BCD

Дж. А. ЛИБЕРМАН (США): Описанное в докладе P/142 детектирующее устройство представляет значительный интерес. Хотелось бы оценить стоимость этой системы по отношению к общей стоимости станции. Какие эксплуатационные меры необходимы для того, чтобы поддерживать эту систему в рабочем состоянии? Как часто система выходит из строя?

Б. ХОЛМС (Соединенное Королевство): Вопрос, по-видимому, основан на предположении о необходимости использования большего количества оборудования. Однако атомная электростанция в Сайзуэлле не требует значительного увеличения оборудования по сравнению с ранее построенными станциями. Капитальные затраты на такое детектирующее устройство колеблются в пределах 1—2% от общей стоимости станции. Основную проблему в такого рода устройствах представляет электронное контрольно-измерительное оборудование, которое составляет лишь небольшую часть подобного оборудования атомной станции. Требования, предъявляемые к оборудованию такой системы, мало отличаются от требований, предъявляемых к значительной части оборудования станции. Однако механическое оборудование является весьма существенным, что связано с загрузочным желобом, и стоимость его эксплуатации должна составлять определенную часть стоимости эксплуатации всего оборудования реактора.

Г. У. ГРИНВУД (Соединенное Королевство): Количество поврежденных тепловыделяющих элементов в реакторах атомных электростанций в Беркли и Брадуэлле Центрального управления электроснабжения Великобритании составляет 1 на 10 000. Тепловыделяющие элементы с магноксовой оболочкой не разрушались из-за кавитации, которая была причиной повреждения используемых раньше тепловыделяющих элементов колдерхоллских реакторов. Некоторые выгруженные из реактора и обследованные после облучения тепловыделяющие элементы указали на необходимость повышения абсолютной чувствительности детектирующего устройства BCD.

Ж. ВЕЙЛЛ (Франция): Приведенная в табл. 1 доклада P/146 чувствительность детектирующей системы BCD, используемой для атомной электростанции в Сайзуэлле, кажется удивительной. Это расчетные или экспериментальные данные? Если это экспериментальные данные, то почему загрязнения на поверхности тепловыделяющего элемента не дают более высокого счета, обусловленного фоном? Можно ли сни-

* Паркер и др. «Параметрические исследования выделения продуктов деления из двуокиси урана». Третья конференция по химии ядерных реакторов. Доклад КАЭ США TID-7641 (1962).

зить количество загрязнений по сравнению с имеющимися до настоящего времени данными порядка 0,01 *мкг/см*²?

В. ХОЛМС (Соединенное Королевство): Эта цифра для загрязнения оболочки тепловыделяющего элемента была использована при расчете чувствительности системы, приведенной в докладе. Указанная цифра основана на экспериментальных данных, точность которых весьма сомнительна. Современные опыты дают более высокое значение. Однако было найдено, что в случае колдерхоллских и чепелкросских реакторов эта неточность компенсируется расхождением между расчетными характеристиками и характеристиками действительной системы. В связи с этим величина счета, обусловленного фоном, все еще базируется на первоначальных расчетах.

Acta de la sesión 3.9

Investigaciones sobre la seguridad nuclear

Presidente: F. R. Farmer (Reino Unido)

Estudios sobre la cinética y los estados transitorios de los reactores

Documento P/282 (presentado por J. A. Lieberman)

DISCUSIÓN *

B. G. DUBOVSKY (URSS): ¿Tiene usted en cuenta al diseñar reactores rápidos, la posibilidad de que aparezca un defecto de reactividad positivo (debido a desplazamiento accidental del refrigerante de sodio y el consiguiente endurecimiento en el espectro de los neutrones) junto con defectos en el sistema de seguridad? ¿Qué precauciones técnicas incluye en el diseño?

J. A. LIEBERMAN (Estados Unidos de América): Creo que es el Sr. McCarthy quien debe responder a esa pregunta.

W. J. MCCARTHY (Estados Unidos de América): La posibilidad de un coeficiente Doppler positivo debido a pérdidas de sodio, fue primeramente señalada por Okrent y Dickerman. Esta posibilidad no se ha tenido específicamente en cuento en el diseño de los reactores de 1 000 MW(e) expecto para tomar las precauciones usuales contra pérdidas de sodio ya sea en general o localmente. En cualquier diseño serio, esta posibilidad tiene que ser considerada con detalle.

B. G. DUBROVSKY (URSS): De acuerdo con su experiencia hasta el presente ¿se puede evaluar la conveniencia de utilizar en los reactores grandes capas protectoras para impedir la diseminación de productos radiactivos en caso de accidente?

J. A. LIEBERMAN (Estados Unidos de América): En los Estados Unidos se han construido y han funcionado satisfactoriamente un cierto número de recipientes de confinamiento para reactores de agua. Se han llevado a cabo pruebas detalladas del ritmo de escape y se han conseguido ritmos diarios de escape del 0,1 % ó menos del volumen confinado. En este sentido le sería de interés el manual sobre confinamiento que va a ser publicado por Cottrell *et al.* del Oak Ridge National Laboratory.

W. HÄFELE (República Federal de Alemania): La evacuación de aire a la atmósfera se realiza en la mayoría de los edificios de confinamiento de los reactores a través de un juego de filtros. Normalmente estos filtros sólo soportan ligeros cambios de presión. ¿Se trabaja actualmente en los Estados Unidos en el desarrollo de filtros resistentes a altas temperaturas y a presión? ¿Cuáles son los límites de temperatura y presión de los mejores filtros disponibles? ¿Cuáles son los correspondientes factores de filtrado?

J. A. LIEBERMAN (Estados Unidos de América): Esta es una cuestión importante. Uno debería saber cómo va a funcionar un sistema de purificación de aire en condiciones de accidente, es decir a temperatura y presión anormales. Los filtros absolutos de aire (tipo papel delgado) actualmente en uso, pueden solamente resistir presiones dentro del margen de varias pulgadas de agua y unos pocos centenares de grados Fahrenheit. Uno de los objetivos importantes del trabajo descrito en nuestra memoria es obtener datos técnicos sobre el funcionamiento de tal sistema en condiciones lo más semejantes posible a las que se presentarían en caso de un accidente serio.

Los dispositivos de purificación de aire generalmente en uso, poseen eficiencias de evacuación de un 99,8 % ó más para partículas de 0,3 μ de diámetro y de alrededor del 99,95 % para iodo, en condiciones normales de operación.

* Véase también la discusión general en esta sesión.

K. A. MAHMOUD (República Arabe Unida): Deduzco de la memoria que su presupuesto para trabajar en seguridad nuclear es de 18 millones de dólares para el año 1964. ¿Qué porcentaje representa esta cifra respecto del presupuesto anual dedicado para trabajos sobre la utilización pacífica de la energía atómica en los Estados Unidos? Me gustaría también conocer cuál es el porcentaje de personal dedicado a problemas de seguridad nuclear.

J. A. LIEBERMAN (Estados Unidos de América): Aunque no poseo cifras determinadas, el porcentaje del presupuesto total que se dedica al desarrollo de la investigación sobre seguridad nuclear es bastante pequeño, probablemente alrededor de un 4%. Sin embargo, aquí no se incluyen trabajos de este tipo que se llevan a cabo en relación con proyectos específicos de reactores, ni trabajos relativos a protección de la salud y control del ambiente.

El número estimado de personas implicadas en el programa descrito está comprendido entre 400 y 500.

R. D. COLLINS (Reino Unido) *: En el documento P/282, en la sección sobre el comportamiento de los productos de fisión, se indica que, en ciertas condiciones, el yodo tiende a reaccionar con impurezas para formar compuestos que no son eficazmente evacuados por los sistemas actuales de purificación de aire. Es cierto que se pueden formar compuestos más difíciles de atrapar que el vapor de yodo, y, tal como se indica en el documento P/186, hemos identificado algunos de ellos, entre los cuales el yoduro de metilo es el más importante. Sin embargo, en un número considerable de experimentos elegidos de modo que incluyesen todas las condiciones de los accidentes postulados para un reactor de diseño original refrigerado por gas, no se encontró ningún compuesto de yodo que no pudiera ser atrapado por una combinación de filtros de alta eficiencia y carbón vegetal, siempre que este dispositivo estuviese diseñado y funcionase adecuadamente. Las primeras fases de una investigación de las condiciones en los reactores de agua tampoco han proporcionado formas nuevas que sean altamente penetrantes.

El vapor de yodo es activo y se adhiere tan firmemente a las impurezas de algunos carbones vegetales que se retiene bien aún cuando otros vapores compitan con él para conseguir espacio sobre la superficie del carbón. El yoduro de metilo es, sin embargo, mucho menos activo; a igualdad de peso de carbón puede quedar retenida mucha menor cantidad del mismo y puede ser desplazada más fácilmente por otros vapores. Es esencial, por tanto, elegir el carbón cuidadosamente y probarlo con la mezcla de gases que se espera vaya a descontaminar; un carbón que es bueno, por ejemplo, para retener el yodo a alta temperatura, no lo es necesariamente para el yoduro de metilo.

* Este comentario no fue formulado durante la sesión.

En el pasado ha habido escasez de experimentos bien documentados que sean apropiados a las condiciones postuladas para accidentes en un reactor y en particular han faltado pruebas en cuanto a la naturaleza y eficiencia de retención de los compuestos de productos de fisión que se forman. Por ejemplo, incluso hoy se sabe poco acerca del escape de dióxido de uranio de la vaina de acero, calentado en vapor a una temperatura superior al punto de fusión de la vaina.

Sin embargo, tal como se refleja especialmente en los documentos P/282 y P/285 existe un interés creciente en tales experimentos y en el desarrollo de un cierto número de técnicas interesantes de clasificación. Espero que no tarde en llegar el día en que los diseñadores de reactores dispongan de datos fidedignos de funcionamiento.

Documento P/382

DISCUSIÓN

W. J. MCCARTHY (Estados Unidos de América): ¿Se ha comprobado experimentalmente el modelo analítico para la moderación de la bomba centrífuga que se describe en la memoria? Dicho con más generalidad ¿qué trabajo experimental se ha llevado a cabo respecto de este estudio teórico?

G. N. POLETAEV (URSS): Se han efectuado experimentos en escala limitada, aunque no aparecen tratados en la memoria. Los resultados experimentales estaban en buen acuerdo con la teoría.

Documento P/283 (presentado por W. E. Nyer)

DISCUSIÓN

F. C. BOYD (Canadá): En reactores de uranio natural, tales como los que utilizan agua pesada, el exceso de reactividad es tan pequeño que no son posibles las grandes excursiones que se estudian en la memoria.

¿Puede usted dar una explicación completa del transitorio destructivo de presión que se estudió en el núcleo del SPERT 1-D (Figura 6) después de la explosión de potencia?

W. E. NYER (Estados Unidos de América): Durante el desarrollo del impulso destructivo de presión, las temperaturas del combustible eran tales que el núcleo interior de las placas se había fundido y extendido por la cara exterior de la envoltura. La tensión mecánica era despreciable y el exterior de las placas estaba por encima de la temperatura de ebullición, de modo que la placa se hallaba prácticamente fundida. En estas condiciones, permanece con su configuración original hasta

que una perturbación produce una rotura del metal líquido y lo dispersa en el agua. Existen varias causas para tal perturbación y cualquiera de ellas pudiera ser la responsable: tensión superficial, tensiones térmicas en las regiones no fundidas de la placa o las vibraciones de los mecanismos del sistema. En algunos experimentos hemos observado que, en ocasiones, las placas se ponen en contacto, atrapando agua y produciendo explosiones locales de vapor de pequeña importancia que fácilmente podían iniciar la rotura de las placas en estado de fundición. Esta rotura y dispersión de gotitas en contacto íntimo con el agua es responsable de la subida repentina de presión. No vemos dificultad en entender este aspecto del impulso destructivo de presión. Para más detalles debe acudir al informe IDO-16884 de Houghtaling, Sola y Spano (publicado por Phillips Petroleum Company, Atomic Energy Division, Idaho Falls, Idaho, 1963).

Documento P/525 (presentado por J. Hulovec)

DISCUSIÓN

W. E. NYER (Estados Unidos de América): Se han efectuado o existen planes para efectuar comparaciones experimentales con los resultados del estudio? En segundo lugar, y en relación a la breve referencia que la memoria hace de ruptura de conductos, ¿ qué tamaños de ruptura han sido considerados en el análisis? ¿ Se ha considerado el caso de un corte completo?

J. HULOVEC (Checoslovaquia): Las propiedades dinámicas de la central serán, naturalmente, sometidas a comprobación experimental. Todavía no tenemos el reactor, pero los generadores de vapor están siendo ya puestos en operación en el soporte de pruebas.

Hemos considerado el caso de corte completo de uno de los doce conductos salientes.

Detección y análisis de fallas

Documento P/436 (presentado por A. Oyama)

DISCUSIÓN

J. M. YELLOWLEES (Reino Unido): La temperatura de ruptura de la vaina (T_d, en la figura 6 de la memoria) debe asociarse con el tiempo tomando como base la ruptura por fluidez. ¿ Qué criterios governaron la selección del tiempo?

Por favor, explique las hipótesis analíticas utilizadas para calcular el transitorio de temperatura de la vaina que aparece en la figura 12. Por ejemplo, la transferencia de calor por radiación entre las partes más calientes y más frías del combustible, puede ser importante cuando la distribución de la asperción es deficiente.

A. OYAMA (Japón): No se ha tratado este punto en la presentación verbal a causa de la limitación de tiempo. Desde luego hicimos ciertas hipótesis para calcular el transitorio de temperatura de la vaina, pero la ruptura de la misma es un fenómeno muy complicado que tendrá que ser investigado con más detalle.

S. LEVY (Estados Unidos de América): ¿ Se ha hecho algún intento para extrapolar este trabajo a sistemas de mayor tamaño? ¿ Se ha prestado atención a las presiones transitorias que aparecen a lo largo del núdeo? En sus pruebas de refrigeración por aspersión ¿ fueron ustedes alguna vez incapaces de refrigerar alguna porción de las barras calientes?

A. OYAMA (Japón): Como respuesta a la primera pregunta debo decir que, por el momento, no tenemos planes para construir este dispositivo experimental a una escala mayor. Nos gustaría primero proseguir los estudios fundamentales con el dispositivo a pequeño tamaño y quizás de aquí a un par de años pueda considerarse el realizarlo a una escala mayor. El Dr. Uchida, coautor de la memoria, está más capacitado para responder a las otras preguntas.

H. UCHIDA (Japón): En cuanto a la segunda pregunta del Sr. Levy, el transitorio de presión en los cuatro canales tiene que determinarse experimentalmente y esto presenta dificultades desde el punto de vista de la precisión; en este sentido encontramos interesante su memoria (P/283).

Sobre la tercera cuestión debo decir que con este sistema de refrigeración por aspersión, es difícil conseguir una refrigeración efectiva de la parte inferior de las barras de combustible caliente. Está en vías de estudio y desarrollo un sistema más eficaz para este reactor.

F. R. FARMER (Presidente): Dr. Lieberman, como portavoz del programa de seguridad nuclear de los Estados Unidos, ¿ quiere añadir alguna cosa en respuesta a las preguntas del Dr. Levy?

J. A. LIEBERMAN (Estados Unidos de América): En el programa estadounidense se ha prestado atención a las caídas de presión a través del núcleo en relación con los estudios para el LOFT (Loss of fluid tests: Pruebas de pérdidas de fluidos). Parece ser que estas caídas de presión que siguen a un accidente con pérdida de refrigerante, pueden producir fuerzas considerables y dar lugar a deformación mecánica, u otros efectos adversos sobre la estructura reticular de soporte, etc. Por consiguiente, sería importante prestar de ahora en adelante más atención a este asunto.

J. E. R. HOLMES (Reino Unido): Voy a hacer un comentario referente a la cuestión de la refrigeración de emergencia por aspersión. Para el reactor SGHW refrigerado por agua ligera hemos encontrado que es beneficioso incluir un sistema de refrigeración de

emergencia para extraer el calor de desintegración. Hemos observado que esto es necesario para asegurar una refrigeración eficiente en todo el intervalo de presiones. Con sistemas de alta calidad, nuestros experimentos han demostrado que los dispositivos de inyección para rociar tienen que ser diseñados cuidadosamente para conseguir una refrigeración eficaz y uniforme. Creo que esto está de acuerdo con las experiencias que ustedes han descrito.

Documento P/517 (presentado por G. Stiennon)

DISCUSIÓN

S. LEVY (Estados Unidos de América): Nos interesaría conocer si se han experimentado algunos de los fallos del tipo considerado en la memoria. ¿Han intentado predecir la temperatura asociada con la inversión de flujo? En caso contrario ¿cree que estos resultados pueden aplicarse a otros reactores?

G. STIENNON (Bélgica): Todavía ho hemos tenido ningún accidente similar a los simulados. Hemos tenido pérdidas de presión pero hasta ahora han ocurrido durante la puesta en marcha cuando la potencia era cero. Consideramos que nuestros cálculos gozan de un margen de seguridad y creemos que se pueden aplicar a otros sistemas diferentes del reactor BR 2. Los estamos aplicando a diferentes configuraciones de este reactor.

D. STOIAN (Bélgica): Como responsable de los cálculos térmicos puedo confirmar esta opinión ya que solamente utilizamos fórmulas convencionales en el estudio de los diferentes sistemas de flujo. En el informe publicado GEX/D/5 aparece información detallada sobre este asunto.

D. F. CARROLL (Sudáfrica): Del texto de la memoria deduzco que se midió experimentalmente el coeficiente de transferencia de calor dentro de un elemento combustible y se comparó a continuación con los valores calculados, tal como se deduce de la figura 5. Si esto es correcto, ¿qué ecuación se utilizó para calcular el coeficiente de transferencia de calor empleado en la figura 5 y cuál es la correlación aproximada entre los valores medidos y los calculados?

G. STIENNON (Bélgica): No se realizaron medidas especiales del coeficiente de transferencia de calor; lo dedujimos a partir de las medidas generales, utilizando las fórmulas convencionales.

DISCUSIÓN GENERAL

Escape y comportamiento de los productos de fisión

F. C. BOYD (Canadá): Se trata de una pregunta relativamente amplia dirigida a los autores de los documentos P/282, P/285, y P/186. ¿Es actualmente suficiente el conocimiento de los fenómenos de transporte y eliminación de productos de fisión, para aplicarlo a los diseños relativos a confinamiento y seguridad general? ¿Pueden los autores hacer algún comentario sobre si estos fenómenos pueden depender o no de que se impida el escape de los productos de fisión al ambiente?

J. A. LIEBERMAN (Estados Unidos de América): La respuesta, en muy pocas palabras, es «no», pero el Sr. Cottrell, que está más familiarizado con este aspecto del programa, puede quizás añadir algún comentario.

W. B. COTTRELL (Estados Unidos de América): La respuesta general e esta pregunta es negativa, aunque hay muchas situaciones específicas en que se conoce el comportamiento de productos de fisión liberados accidentalmente. Hasta la fecha el programa relativamente extenso de los Estados Unidos ha identificado las cantidades de los distintos isótopos que pueden liberarse en una amplia gama de condiciones de accidendentes. Sin embargo, el transporte de estos productos de fisión en sistemas de confinamiento es también función de las estructuras físicas y químicas de la actividad liberada, así como de las características del sistema de confinamento y de las condiciones ambientales. Los estudios sobre la caracterización de los productos de fisión y su comportamiento bajo tales circunstancias, forman parte del programa de seguridad nuclear que se lleva a cabo en el Oak Ridge National Laboratory, donde diferentes aspectos del problema se examinan en un cierto número de instalaciones que incluyen la planta piloto de seguridad. Tales estudios serán ampliados con el experimento sobre sistemas de confinamiento de Hanford y las pruebas de pérdida de flujo de Phillips Petroleum (NRTS). En el documento P/285 se analizan más detenidamente estos programas y los resultados encontrados hasta la fecha.

R. D. COLLINS (Reino Unido): Nosotros no sabemos cómo diseñar sistemas que impidan adecuadamente el escape de cualquier producto de fisión, pero creemos que, al menos en un sistema (el reactor refrigerado por gas, de nueva concepción), hemos identificado las principales dificultades. Hemos obtenido las condiciones que deben cumplirse en el diseño y operación y hemos demostrado que se puede confiar en un tratamiento eficaz del gas.

F. R. FARMER (Presidente): Dr. Lieberman, ¿desea añadir algo aquí sobre el punto de vista de los Estados Unidos? En primer lugar me ha sorprendido un poco su rotunda negativa. Creo que la mayoría de las preguntas en cuestiones de seguridad exigirían una respuesta negativa en un sentido absoluto, pero pienso que existe un grado creciente de confianza en un cierto número de campos.

J. A. LIEBERMAN (Estados Unidos de América): No pretendía que la respuesta fuese categórica. No creo

que haya una ignorancia completa en esta esfera. Los esfuerzos que se llevan a cabo en varios países para caracterizar la naturaleza y el comportamiento de los productos de fisión que pueden desprenderse del combustible de un reactor, proporcionan una información apreciable. Sobre esta base se pueden estimar las consecuencias de un posible accidente en un determinado tipo de reactor con mayor confianza que en el pasado y además los resultados se pueden aplicar al diseño de sistemas de confinamiento o de purificación de aire.

F. C. BOYD (Canadá): No me gustaría dejar una impresión errónea sobre la actitud canadiense o sobre sus conocimientos en cuanto a los sistemas de confinamiento. Creemos que para cada tipo posible de reactor se puede estimar la cantidad de productos de fisión que van a ser liberados con detalle suficiente para permitir el diseño de características especiales de confinamientos adecuados a ese tipo particular.

W. B. COTTRELL (Estados Unidos de América): En el documento P/186 se indica que la proporción de productos de fisión que escapan del combustible, era considerablemente mayor que la encontrada por Parker * en 1962 para escape en un gas inerte. El escape de productos de fisión es función del tiempo, de la temperatura y de la dosis de radiación. El trabajo experimental más reciente de Parker sobre liberación de productos de fisión (parte del cual se cita en el documento P/285) ha dado por resultado escapes superiores a los publicados en el documento de 1962 y son consistentes con el valor de un 30–50% de yodo indicado en el documento P/186 (véase en particular la figura 1 en el documento P/285).

En conjunto, el documento P/186 representa un trabajo excelente. Me gustaría saber más acerca de las singulares propiedades físicas del carbón 207B que le hacen tan adsorbente para el yoduro de metilo.

R. D. COLLINS (Reino Unido): No somos capaces de especificar las características que le hacen descomponer eficazmente el yoduro de metilo.

W. J. MCCARTHY (Estados Unidos de América): ¿Se ha emprendido en el Reino Unido algún trabajo con el fin específico de estimar los escapes de productos de fisión en sistemas de refrigeración a base de sodio o una aleación de sodio-potasio?

R. D. COLLINS (Reino Unido): Actualmente, no tenemos en marcha ningún experimento de importancia sobre la retención de productos de fisión en sistemas con sodio.

* Parker, G. W. *et al.*, *Parametric Studies of Fission Product Release from UO_2 Fuels* (Estudios paramétricos de la liberación de productos de fisión en combustibles de UO_2. Tercera Conferencia sobre Química de Reactores Nucleares, informe USAEC, TID-7641 (1962).

La oxidación del grafito y la seguridad de los reactores

W. J. MCCARTHY (Estados Unidos de América): Se ha indicado en el documento P/185 que la reactividad del grafito moderador frente al oxígeno aumentaba al irradiar con neutrones. ¿Se ha hecho alguna estimación de la dependencia de este aumento en reactividad respecto de la dosis total, es decir, sería la reactividad doble para una dosis doble o hay un efecto de saturación?

J. R. BEATTIE (Reino Unido): En el documento P/185 se establece que el efecto de la dosis de neutrones es aproximadamente de 0,4 μg/gh a 1 000 MWd/t ady. y se trata de un efecto aproximadamente lineal dentro de lo que se conoce actualmente. Se cree que el efecto de la irradiación sobre la reactividad es una combinación de dos efectos, uno a consecuencia de la dosis de neutrones y el otro debido a corrosión radiolítica por CO_2. Tal como se indica en la memoria, se cree que el máximo efecto de esta última causa es un incremento cuádruple en reactividad. El trabajo prosigue con el fin de aclarar ambos efectos y proponemos relacionarlo con medidas hechos sobre grafito utilizado en reactores a medida que la exposición a la radiación continúa.

Sistemas para la detección de cartuchos reventados

J. A. LIEBERMAN (Estados Unidos de América): La discusión del dispositivo de detección que se da en el documento P/142 es muy interesante. Agradeceríamos alguna indicación sobre el porcentaje en que este sistema contribuye al coste total de la central. ¿Qué equipo de mantenimiento se necesita para mantener el sistema en operación? Y finalmente, ¿cuál es la frecuencia con que fallan las vainas?

B. HOLMES (Reino Unido): La pregunta parece basarse en la hipótesis de que se necesitan grandes cantidades de equipo. Sin embargo, la central de Sizewell no requiere un aumento apreciable en equipo si se compara con centrales más antiguas. El coste capital del dispositivo para la detección de cartuchos reventados es del orden de entre un 1% y un 2% del coste de la central. En cuanto a mantenimiento, el mayor problema corresponde al equipo electrónico de instrumentación y control que, en realidad, sólo constituye una pequeña parte del equipo de este tipo que lleva la central. Por consiguiente, en esta fase no se pueden considerar aisladamente las necesidades de personal para mantenimiento del sistema para la detección de cartuchos reventados. Igualmente, el equipo mecánico es esencialmente el asociado con los conductos de carga y su mantenimiento formará parte del equipo general del reactor.

G. W. GREENWOOD (Reino Unido): En los reactores CEGB de Berkeley y Bradwell en el Reino Unido, el

número de elementos combustibles que han registrado fallos ha sido alrededor de 1 en 10 000. En las vainas del Magnox no hay hasta la fecha indicación de fallos en los elementos combustibles debidos a cavitación que fue la causa de fallos en algunos elementos combustibles de los reactores de Calder Hall. Algunos elementos combustibles que han sido retirados de operación han demostrado estar en buenas condiciones al ser examinados, lo que sugiere que poco se puede ganar aumentando la sensibilidad absoluta del dispositivo de detección.

J. WEILL (Francia): Las sensibilidades para la detección de cartuchos reventados que aparecen en la tabla 1 del documento P/142, especialmente en el caso de la central de Sizewell, parecen sorprendentes. ¿Se obtuvieron por cálculo o experimentalmente? En el último caso, ¿cómo es que la contaminación superficial del elemento combustible no da un mayor recuento de fondo? ¿Se ha podido reducir esta contaminación comparada con los resultados obtenidos hasta ahora, que eran del orden de 0,01 $\mu g/cm^2$?

B. HOLMES (Reino Unido): La cifra sugerida por el Sr. Weill para contaminación de la vaina ha sido la usada en el cálculo de sensibilidad citado en la memoria. Esta cifra se basó originalmente en un valor experimental, cuya exactitud se ha puesto en duda posteriormente. Experimentos recientes han sugerido una cifra más alta. Sin embargo, se ha encontrado, que en el caso de los reactores de Calder Hall y Chapelcross, este error estaba anulado por una discrepancia en la correlación entre el funcionamiento real del sistema y la que había sido predicha. Por esta razón, el recuento de fondo se ha basado todavía en los cálculos originales.

Session 3.8

CHEMICAL PROCESSING PLANT SAFETY; FUEL TRANSPORT

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Eléments de sécurité tirés de six années d'exploitation de l'usine de traitement de combustibles irradiés du Centre de Marcoule

par J. Rodier *

Dans une usine de traitement de combustibles irradiés comme dans toute autre usine, il faut veiller sans relâche à la continuité de la production. Tout incident, aussi bénin soit-il, peut avoir des répercussions sur la productivité. Ainsi, en considérant l'ensemble des conséquences d'un accident, il apparaît que la continuité de la production ne peut être obtenue sans tenir compte de la protection des travailleurs. « Sécurité-travailleurs » et « Sécurité-production » ne sont que les volets d'un même diptyque. Ces deux facteurs sont inséparables et il faut s'efforcer de les faire progresser parallèlement.

Une bonne conception des installations n'est pas suffisante pour éviter les incidents de fonctionnement; elle doit être doublée par une étroite collaboration entre l'homme et la technique. En règle générale, l'accident provient de causes qui peuvent être non seulement techniques, mais aussi humaines. La défaillance d'un matériel ou d'une méthode trouve souvent son origine dans le comportement des individus et les conséquences d'un incident peuvent être considérablement modifiées par des initiatives heureuses ou malheureuses. Il en découle que l'éducation du personnel, le respect des consignes de travail, sont des éléments importants dans la sécurité d'une usine. Pratiquement, il est indispensable qu'une prévention psychologique, cherchant à éliminer les causes d'accidents liées au comportement de l'homme, vienne compléter la prévention technique.

La sécurité d'une usine de traitement de combustibles irradiés dépend ainsi d'un ensemble complexe d'éléments techniques et d'éléments psychologiques étroitement liés.

ÉLÉMENTS TECHNIQUES

Il est bien évident que dans une usine de traitement des combustibles irradiés, il est en premier lieu nécessaire de respecter les *impératifs classiques de la sécurité* dans les usines chimiques. Ainsi par exemple, un incendie des filtres d'entrée de la ventilation de l'usine, provoqué par des projections de soudure, nous a montré les dangers présentés par les filtres inflammables. A la suite de cet incident, la décision a été prise de ne monter dans les circuits de ventilation que des éléments filtrants incombustibles. Toutefois, par leur présence tout au long de la chaîne du traitement, les produits de fission ou le plutonium rendent insuffisants les éléments de sécurité classiques. Ils imposent que des *éléments de sécurité radiologique* leur soient superposés.

Les principes généraux de la prévention technique contre les risques radioactifs sont maintenant bien connus. Il est bon toutefois de souligner que pour lutter contre les dangers d'irradiation externe, l'emploi de blindages convenables, réalisés dans beaucoup de cas en éléments aisément démontables et récupérables, permet de limiter les doses d'irradiation du personnel beaucoup plus sûrement que la pratique du travail à distance ou la limitation de la durée d'exposition. De la même manière, il est préférable que la prévention des risques de contamination radioactive repose sur le confinement plutôt que sur l'élimination *a posteriori* des produits dispersés.

Du point de vue technique, la sécurité radiologique dépend essentiellement de la bonne conception de l'appareillage et des locaux.

Conception de l'appareillage

D'une façon générale, un soin particulier doit être apporté au choix du matériel. Les réalisations doivent être orientées par la recherche d'une grande robustesse, la possibilité d'une surveillance aisée des différents organes et la facilité des réparations. La robustesse du matériel et la sûreté de son exploitation seront atteintes par une grande simplicité, aussi bien dans les formes générales que dans la mécanique.

L'expérience montre que les tuyauteries et la chaudronnerie en général sont souvent le point faible de toute installation chimique. Leur défectuosité peut, dans l'industrie nucléaire, être la cause de graves inconvénients. Le maximum de sécurité doit donc être pris contre les phénomènes de corrosion. A Marcoule, des

* Commissariat à l'énergie atomique. Avec la collaboration de C. Jouannaud, E. Mazaury, Y. Fruchard et P. Regnaut.

aciers inoxydables particulièrement résistants ont été adoptés pour la réalisation des appareils les plus sensibles: cuves de dissolution et évaporateurs de produits de fission. Il est toutefois à remarquer que le choix judicieux des matériaux employés ne servirait à rien si par la suite il n'était pas apporté un grand soin au contrôle de tous les lots de tubes ou de tôles et si les réalisations du constructeur n'étaient pas suivies de très près par les visites fréquentes d'un contrôleur.

Les soudures constituent toujours les points névralgiques d'un réseau de tuyauteries. Les techniques d'assemblage sont étudiées à l'échelle du laboratoire: composition des métaux d'apport et de base, structure métallographique. Les soudures strictement indispensables sont à réaliser avec le plus grand soin par des soudeurs expérimentés dont les connaissances et la dextérité sont vérifiées par des examens périodiques. Le contrôle radiographique des soudures doit constituer une règle; des essais de ressuage au rouge organol peuvent ensuite parfaire les contrôles.

Malgré tous les soins apportés dans la construction d'un matériel, il est inévitable que des défauts de fonctionnement se manifestent après un temps de service plus ou moins long. La remise en état d'un matériel traitant des solutions fortement radioactives est, la plupart du temps, une opération longue et difficile. Il est donc nécessaire, pour assurer dans tous les cas la sécurité-production de disposer d'*unités de secours* qui peuvent être mises en fonctionnement immédiatement en cas de défaillance de l'unité en service.

Des réparations peuvent être envisagées dans les enceintes fortement radioactives à condition que cette éventualité ait été prévue. Pour qu'elles se déroulent dans de bonnes conditions, il est nécessaire de disposer de moyens d'observation directe ou indirecte des défauts (hublots, périscopes, télévision) et de moyens d'évaluation des dangers courus par les travailleurs. Il est surtout indispensable de pouvoir ramener ces dangers à des niveaux acceptables.

Ainsi, à Marcoule où la partie « haute activité » de l'usine a été conçue sur le principe de l'*intervention directe*, l'entretien ou le remplacement des appareils ne peut s'effectuer qu'après décontamination. Cette décontamination doit être très poussée pour que les doses d'irradiation intégrées par les travailleurs restent dans des limites raisonnables. L'expérience a montré qu'il était indispensable d'arroser énergiquement les parties actives avec des solutions décontaminantes; les éjecteurs à vapeur sont d'une grande utilité pour de telles opérations. Par ailleurs, il peut être bon de procéder périodiquement à la décontamination de certains circuits afin d'éliminer des points d'accumulation éventuels et rendre ainsi plus aisées les opérations de décontamination poussée imposées par les interventions.

Quel que soit le degré de décontamination atteint, le danger d'irradiation externe, dans une intervention d'entretien ou de réparation, n'est jamais totalement annulé. Pour que, malgré tout, l'intervention se déroule dans de bonnes conditions, il est souhaitable que des blindages isolent les appareils actifs les uns des autres. Dans l'usine, la séparation des appareillages est réalisée au moyen de cellules indépendantes en béton dont un des côtés est constitué de briques en béton baryté dense, assemblées par simple emboîtement.

Conception des locaux

Une bonne conception des locaux est utile à la sécurité-production. Dans le domaine de la sécurité-travailleurs elle devient indispensable. Elle est en particulier la condition essentielle d'une lutte efficace contre la contamination radioactive.

La prévention des risques de contamination radioactive repose toujours sur le confinement et l'élimination des produits contaminants. Cela implique l'établissement de « zones » différentes selon le gradient du risque, avec des conditions d'accès et de travail réglementées.

Dans les circonstances normales, les personnes ne peuvent accéder à une zone donnée qu'à partir de la zone à risque immédiatement inférieur après avoir subi un contrôle et respecté des obligations vestimentaires. A l'entrée d'une zone à risque permanent de contamination, la tenue de travail normal est complétée par un survêtement spécial en toile. L'expérience nous a en effet montré qu'il était plus facile de « peler » les travailleurs à la sortie d'une zone contaminée que de leur imposer un contrôle long, fastidieux et toujours moins efficace.

Le complément indispensable du cloisonnement d'une usine est une ventilation bien conçue. Indépendamment du taux de renouvellement de l'air dans les locaux, l'efficacité de la ventilation repose essentiellement sur le maintien permanent d'une dépression suffisante entre les différentes zones. L'expérience montre que dans ce domaine les problèmes de ventilation se ramènent souvent à des problèmes de portes dont on oublie trop souvent, en les laissant ouvertes, que le rôle fondamental est de clore une ouverture.

Problèmes particuliers liés à la criticité

La sécurité vis-à-vis des risques de criticité dans une usine de traitement des combustibles irradiés peut reposer, soit sur une limitation de la masse de plutonium, soit sur une limitation de la concentration des solutions, soit sur un choix judicieux des formes et des dimensions de l'appareillage. Ce dernier mode de contrôle doit être recherché chaque fois que cela est possible. Une combinaison de ces modes de contrôle peut parfois être utilisée.

Quoique conçue à une époque où les techniques assurant le contrôle de la criticité étaient encore mal

connues, l'usine a le mérite d'avoir apporté une certaine expérience en la matière. Si aucun accident de criticité n'a été à déplorer jusqu'ici, cela est dû avant tout à la rigoureuse observation des consignes par le personnel d'exploitation. Toutefois, pour limiter le nombre des consignes et diminuer les risques d'erreurs dues aux facteurs humains, nous cherchons à développer le plus possible les appareillages et les circuits géométriquement sûrs. Actuellement des améliorations notables sont en cours. Les études nécessitent l'emploi de méthodes de calcul assez poussées (codes de transport, codes Monte-Carlo) et l'implantation des appareils doit être étudiée avec soin de façon à tenir compte des phénomènes d'interaction.

D'ici quelques années, l'usine sera pratiquement totalement sûre par la géométrie depuis l'extraction II * jusqu'à la fluoration du plutonium (l'exception concerne les réservoirs d'effluents). Cependant, dans la partie de l'usine traitant des volumes importants de solutions à des concentrations relativement faibles, la sûreté ne peut que reposer sur un contrôle très strict de la concentration. Les analyses nécessaires sont effectuées en laboratoire; elles sont doublées par des appareils de contrôle en continu (compteurs α, compteurs neutrons).

Le service d'exploitation est responsable de la sécurité en matière de criticité.

L'expérience de ces dernières années a montré l'importance qu'il y a à se préoccuper des questions de criticité dès le stade de la conception de l'usine, de façon à étendre au maximum le domaine de la sécurité par la géométrie. La recherche d'une telle sécurité ne peut cependant qu'alléger le nombre des consignes, sans les supprimer tout à fait: celles-ci doivent être simples et parfaitement assimilées par le personnel. Enfin, il est très important d'informer le personnel des risques possibles, de façon à développer chez lui des réflexes préventifs, particulièrement lors d'opérations non routinières.

FACTEURS PSYCHOLOGIQUES

Les facteurs techniques de sécurité qui viennent d'être étudiés ne sont pas à eux seuls suffisants. Quels que soient la qualité du matériel et le degré d'automatisme atteint, la continuité de fonctionnement d'une usine implique l'intervention de l'homme qui surveille, dépanne, prend des initiatives. La sécurité technique n'est jamais définitive et à tout instant elle a besoin d'être contrôlée. Elle ne peut rester efficace qu'avec la collaboration des travailleurs. Il est donc nécessaire qu'une prévention psychologique vienne compléter la prévention technique. Cette prévention psychologique repose sur l'éducation du personnel, l'organisation du travail et la définition des consignes de sécurité, le contrôle du respect des consignes et du comportement des travailleurs et enfin le rappel incessant de la nécessité de la prévention des risques radioactifs. Le démarrage des installations a eu lieu avec du personnel qui, dans sa grande majorité, travaillait pour la première fois en milieu radioactif. Rares étaient les agents qui possédaient une formation suffisante en matière de radioprotection. Un grand nombre d'entre eux conservait une bonne part des préjugés rencontrés dans le grand public. Ils étaient souvent pétris de crainte ou d'inconscience. Seule l'éducation de la masse des travailleurs pouvait remédier à cette situation. Pour démontrer l'importance des règles d'hygiène radioactive et faire admettre la nécessité de consignes de sécurité très strictes, des cycles de conférences ont été organisés de façon à exposer aux travailleurs les objectifs à atteindre en matière d'hygiène radioactive. En outre, des brochures simples, illustrant les points importants et les idées directrices, sont distribuées à tous les agents et l'attention du personnel est constamment sollicitée par des affiches éditées bimensuellement qui rappellent les diverses consignes et la signification des symboles conventionnels.

Grâce à l'ensemble de ces moyens audio-visuels, la grande masse des travailleurs a pris conscience des risques réels que présente l'industrie nucléaire. L'expérience montre que la sécurité en a largement bénéficié.

ORGANISATION DU CONTRÔLE RADIOLOGIQUE

L'importance des risques potentiels pouvant entraîner des accidents graves et la perte d'investissements énormes justifient une réglementation très stricte des méthodes de travail. Des consignes précises doivent fixer les procédures d'exploitation. Elles doivent ensuite être complétées par des consignes réglementant les conditions de travail en milieu radioactif. Il est en effet nécessaire qu'un contrôle très strict des dangers d'irradiation externe ou de contamination auxquels sont exposés les travailleurs soit exercé pendant les opérations en présence de radioéléments.

Au Centre de Marcoule, le choix a été fait de confier ce contrôle à des agents spécialisés appartenant à un service de radioprotection indépendant des services d'exploitation. Cette organisation présente plusieurs avantages. Tout d'abord elle décharge les équipes d'exploitation de la lourde tâche que représente l'évaluation et la surveillance de l'évolution des différents risques radioactifs présentés par l'installation; le contrôle de l'irradiation et de la contamination éventuelle des agents à la fin du travail, et le rassemblement des résultats des mesures radiologiques.

La séparation des équipes de radioprotection et d'exploitation évite par ailleurs, dans bien des cas, que l'idée « production avant tout » ne prenne trop aisé-

* Dans le procédé utilisé à Marcoule, l'extraction II correspond au passage en phase aqueuse du plutonium extrait par solvant en même temps que l'uranium lors de l'extraction I.

ment le pas sur la sécurité-travailleurs. Elle garantit que toute intervention, aussi urgente et délicate soit-elle, sera conduite en tenant un compte exact des risques qu'elle entraîne.

Ces avantages très réels trouvent évidemment leur contrepartie dans quelques inconvénients. Le premier réside dans le fait que les équipes d'exploitation, déchargées de tout souci de radioprotection, ne sont pas toujours très au courant des méthodes d'évaluation des risques radioactifs. Elles risquent, prises au dépourvu, de prendre des initiatives plus ou moins malheureuses; ces cas sont toutefois fort rares. Le deuxième désavantage peut être trouvé dans la lourdeur du système. En fait, l'expérience montre que les liaisons entre radioprotection et exploitation peuvent être assez souples pour que les travailleurs soient toujours informés rapidement des risques qu'ils courent et des précautions qu'ils doivent prendre. Il suffit pour cela que des informations mutuelles puissent être échangées à tous les échelons.

L'évaluation correcte des risques radioactifs créés par une installation doit reposer, d'une part sur des mesures faites par un réseau de détecteurs fixes contrôlant les débits de dose d'irradiation ou la contamination atmosphérique en divers points des locaux et, d'autre part, des observations faites au cours de rondes routinières et inopinées effectuées par les agents de radioprotection munis d'appareils portatifs. La fréquence des contrôles doit être dictée par l'importance du risque, mais il faut s'attacher à ce qu'elle soit toujours la plus grande possible. L'efficacité de la surveillance, la rapidité de la détection, c'est-à-dire le degré de la sécurité-travailleurs sont en effet étroitement liées au nombre de mesures.

Afin de pouvoir exploiter efficacement les résultats des contrôles de radioprotection, notre expérience a par ailleurs montré qu'il était utile de les grouper sur des fiches qui permettent à la fois leur diffusion et leur conservation en archives. Différents types de fiches correspondant aux principaux aspects de l'exploitation sont utilisées:

Fiches de salubrité radioactive d'un lieu de travail rassemblant les résultats des mesures radiologiques.

Fiches de travaux de caractères exceptionnels. Ces travaux sont définis à Marcoule comme des travaux au cours desquels les travailleurs sont exposés à des débits de dose supérieurs à 7,5 mrem/h. L'établissement d'une fiche avant chacun d'eux permet de définir les risques courus, les moyens de protection convenables et, surtout, de suivre le personnel pendant les périodes les plus dangereuses de son travail.

Fiches d'interventions de décontamination dont le nombre est un renseignement précieux sur le risque radioactif présenté par l'installation.

Fiches de contamination corporelle accompagnant les agents contaminés envoyés au bloc médical pour examen et décontamination.

RÉSULTATS DES CONTRÔLES RADIOLOGIQUES

L'exploitation statistique des fiches de radioprotection nous permet d'avoir une idée précise du fonctionnement des installations et du danger radioactif auquel les travailleurs sont exposés. Les fiches de travaux à caractère exceptionnel montrent par exemple que, dans le cas de l'usine de traitement de combustibles irradiés du Centre de Marcoule, le nombre annuel d'autorisations de dépassement de débit de dose accordées de 1959 à 1963 est passé de 100 à 1 500. La plupart d'entre elles correspondent à des débits de dose compris entre 7,5 mrem/h et 75 mrem/h. Le pourcentage d'autorisations accordées au-delà de 75 mrem/h a été de 24 % en 1959 et de 1 % en 1963. Dans le même intervalle, le nombre d'agents-jours correspondant à ces opérations est passé de 1 200 à 10 800. Le phénomène découle du rythme de l'exploitation mais aussi des travaux d'aménagement effectués pour améliorer les caractéristiques de l'usine.

Il est toutefois à noter que, malgré cette augmentation, l'exposition radioactive des agents est restée faible. En effet, aux 1 500 travaux à caractère exceptionnel correspondent 7 300 heures de travail, soit seulement 0,5 % du nombre total des heures de travail effectuées par le personnel de l'usine. Par ailleurs, les temps journaliers de participation à ces opérations sont généralement courts. Comme le montre la figure 1, ils ne sont que dans 10 % des cas supérieurs à 1,5 heure. Ceci est dû pour une part à la nature des tra-

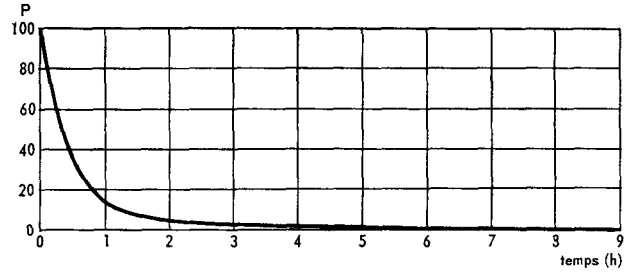


Figure 1. Répartition des temps journaliers de participation aux manipulations à caractère exceptionnel effectuées en 1963.
P = % des temps journaliers supérieurs à une valeur de t donnée.

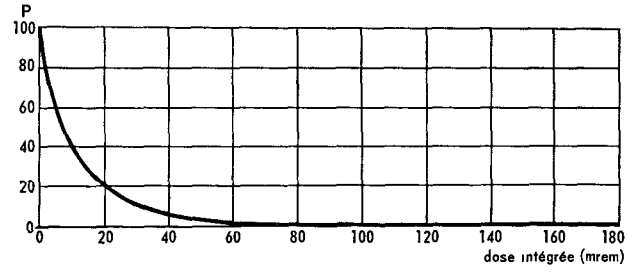


Figure 2. Répartition des doses journalières intégrées au cours des manipulations à caractère exceptionnel effectuées en 1963.
P = % des doses journalières supérieures à une valeur de d donnée.

vaux effectués mais surtout à la rotation du personnel et à la limitation du temps d'exposition de chaque agent.

D'après la figure 2, les valeurs des doses d'irradiation journalières reçues par le personnel ayant participé aux travaux à caractère exceptionnel restent inférieures à 10 mrem pour 60 % des agents et à 35 mrem pour 90 % d'entre eux. La valeur maximale enregistrée est de 180 mrem. Compte tenu des doses intégrées et des temps d'exposition, le débit de dose moyen auquel ont été exposés les travailleurs pendant les travaux à caractère exceptionnel de 1963 apparaît voisin de 15 mrem/h.

Tous ces résultats font la preuve que l'exploitation et l'entretien d'une usine de traitement de combustibles irradiés sont compatibles avec une exposition des travailleurs à des doses ou des débits de dose très faibles.

L'exploitation des fiches de contamination corporelle révèle que, depuis 1959, le nombre de cas observés dans l'usine de Marcoule a peu varié. Il y a en moyenne 170 cas pas an qui se répartissent de la façon suivante: 10 à la Station de traitement des effluents, 40 au laboratoire et 120 à l'usine proprement dite. Grâce à la discipline de travail exigée des agents, le nombre de cas de contamination corporelle apparaît très faible. Il ne correspond en effet qu'à un taux moyen de fréquence égal à 0,5 (cas pour 10 000 h de travail). La plupart des cas sont évidemment enregistrés dans les zones à risque permanent de contamination (60 %) et dans les zones occasionnelles (26 %).

Les parties du corps les plus fréquemment contaminées sont, d'une part, les mains et les membres supérieurs souvent en contact avec des produits contaminants et, d'autre part, la tête (face et cheveux) souvent mal protégée contre les projections accidentelles. La figure 3 indique la répartition de ces différents cas de contamination pour l'année 1963 tandis que la

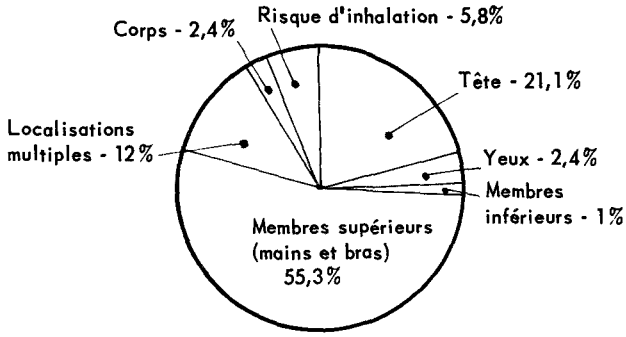


Figure 3. Répartition des cas de contamination corporelle en fonction de leur nature

figure 4 montre leur répartition en fonction de la nature et de l'état physique des radioéléments contaminants.

Par ailleurs, de l'exploitation des fiches d'intervention de décontamination, il ressort que la décontamination des locaux et de l'appareillage de l'usine ne nécessite qu'un nombre d'agents spécialisés relativement faible. L'effectif des décontamineurs correspondant à un emploi à plein temps est en effet sensiblement égal à 3 % du personnel d'exploitation.

Les efforts dépensés pour obtenir une sécurité maximale trouvent leur récompense dans la faible exposition des travailleurs aux radiations. Les résultats fournis, tant par le contrôle des doses d'irradiation externe que par le contrôle de la contamination interne, sont dans ce sens très significatifs:

Irradiation externe du personnel

Les doses d'irradiation externe des agents sont contrôlées au moyen des films dosimètres individuels dont le port est obligatoire dans toute zone contrôlée.

La figure 5 montre l'évolution des irradiations trimestrielles moyennes, pour la période s'étendant du

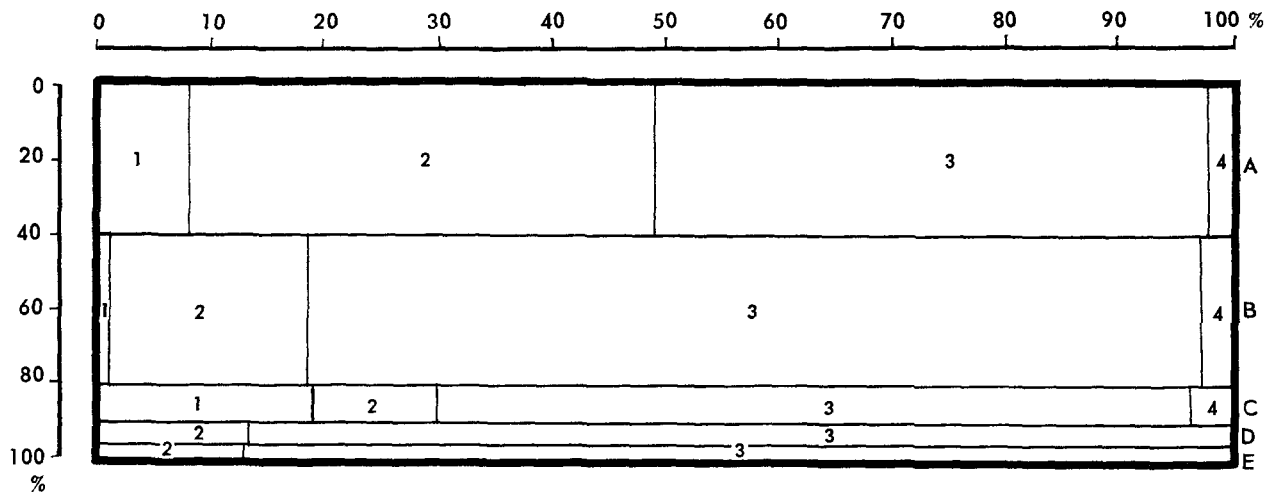


Figure 4. Répartition des cas de contamination corporelle enregistrés en 1963 en fonction de la nature du radioélément contaminant et de son état physique

Etat physique: A: Solution aqueuse; B: Solide; C: Gaz; D: Lubrifiant; E: Solvant; Nature du radioélément: 1: Uranium; 2: Plutonium; 3: Produits de fission; 4: Divers

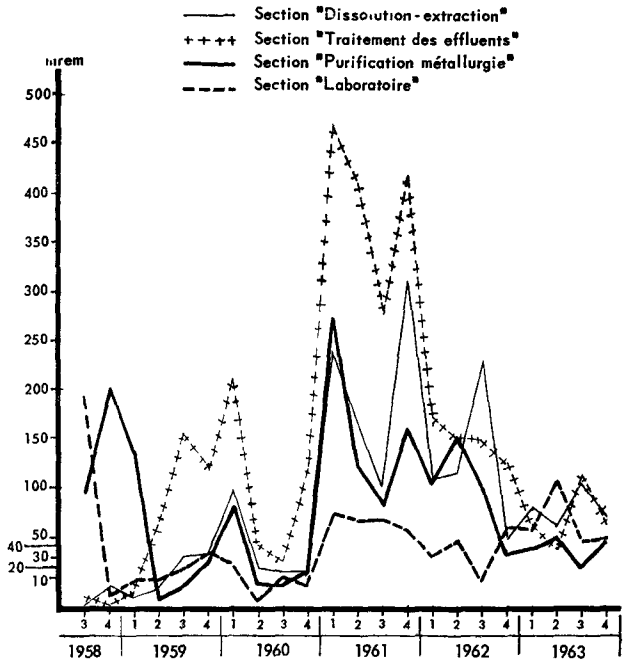


Figure 5. Irradiation moyenne trimestrielle d'un agent (dose réelle en profondeur).

1er janvier 1958 au 31 décembre 1963, d'un agent appartenant respectivement à la Section dissolution-extraction, la Section purification-métallurgie, la Section laboratoire, la Section traitement des effluents.

Il apparaît que pour la Section traitement des effluents l'irradiation moyenne des agents a progressivement augmenté jusqu'en 1961. A partir de cette date, la diminution des doses reçues par le personnel a été obtenue par la mise en place de blindages convenables sur certaines conduites et certains bassins contenant des effluents particulièrement actifs. Le graphique correspondant à la Section dissolution-extraction montre une évolution semblable.

En fait, l'année 1961 correspond à Marcoule à un tournant dans le domaine de la radioprotection. La mise en place de protections complémentaires, une discipline radioactive encore plus stricte ont eu une influence énorme sur le degré de sécurité de l'usine.

Contamination interne du personnel

La surveillance de la contamination interne portée par les travailleurs est assurée par des analyses des produits d'excrétion et par des examens spectrométriques α. Ces examens sont effectués d'une façon systématique à intervalles réguliers. Ils sont complétés par des mesures localisées, des analyses de matières fécales, etc., lors des incidents où il existe une présomption de contamination interne.

Contamination par le plutonium – L'ensemble des résultats des six dernières années montre qu'aucune élimination urinaire stable, supérieure à 0,1 pCi par 24 h n'a pu être mise en évidence chez les personnes surveillées. Ceci permet de vérifier l'absence d'une contamination chronique de niveau significatif. Un certain nombre de contaminations isolées ont été cependant décelées, soit à la suite d'incidents, soit grâce aux analyses urinaires systématiques. Elles sont groupées dans le tableau 1 où elles sont exprimées en pourcentage de la quantité maximale admissible dans l'organisme (0,04 μCi).

Tableau 1. Surveillance de la contamination par le plutonium

Années	Analyses effectuées	Cas de contamination			Total des cas de contamination
		Inférieurs à 1%[a]	Entre 1% et 5%*	Entre 5% et 10%*	
1958–1959	971	11	18	1	30
1960	2 045	20	31	1	52
1961	2 012	10	27	2	39
1962	2 023	3	4	0	7
1963	2 031	7	7	1	15
Totaux	9 082	51	87	5	143

[a] Après déduction de l'activité γ naturelle de l'individu.
* % de la QMA (= 0,04 μCi).

Contaminations par les produits de fission – De même, pour les produits de fission, aucune contamination interne significative obtenue par un ensemble cohérent de mesures, n'a pu être mise en évidence au cours des six dernières années. Les résultats globaux des examens mis en œuvre pour surveiller la contamination interne par les produits de fission sont groupés dans le tableau 2.

Tableau 2. Surveillance de la contamination par les produits de fission

Années	Analyses effectuées			Cas de contamination		
	Urines	Spectro γ	Urines positives	Activité γ du corps humain[a]		
				inférieure à 3.10^{-2} μCi	de 3.10^{-2} à $1{,}5.10^{-1}$ μCi	de $1{,}5.10^{-1}$ à 3.10^{-1} μCi
1958–1959	1 059	—	21	—	—	—
1960	1 675	153	15	18	4	0
1961	2 420	400	3	48	3	1
1962	2 271	416	3	40	4	0
1963	2 386	437	8	32	6	0
Totaux	9 811	1 406	50	138	17	1

[a] Après déduction de l'activité γ naturelle de l'individu.

Les urines positives correspondent à une concentration en β totaux (potassium exclu) égale ou supérieure à 100 pCi/l. La majorité des résultats gravite autour de cette valeur, aucune élimination n'ayant été suffisante pour être suivie lors des examens de contrôle ultérieurs.

Les spectres γ du corps humain révèlent tous une nette prédominance de ^{95}Nb-^{95}Zr. Ils montrent toutefois que les contaminations sont en majorité très faible, surtout si l'on tient compte de leur surestimation

systématique, due à l'influence considérable sur la mesure de traces infimes de contamination externe rebelles au lavage.

CONCLUSION

Les quelques incidents survenus à l'usine de traitement de combustibles irradiés du Centre de Marcoule tels que l'incendie des filtres d'entrée de la ventilation de l'usine, le désamorçage d'un siphon avec contamination du circuit de vide, une fuite sur une tuyauterie alimentant les batteries d'extraction et le débordement d'une cuve de dissolution, n'ont jamais eu de répercussion importante dans le domaine de l'exploitation. Les cadences prévues ont toujours pu être maintenues.

Par ailleurs en analysant les résultats obtenus après six années de fonctionnement, il apparaît immédiatement que c'est une erreur de croire que le travail dans l'industrie chimique nucléaire est plus particulièrement dangereux. Qu'il s'agisse en effet d'accidents ou de risques pour la santé du personnel, nos statistiques montrent que l'exploitation d'une usine de traitement de combustibles irradiés peut être conduite de manière très sûre.

La sécurité, et il s'agit aussi bien de la sécurité-production que de la sécurité-travailleurs repose sur une bonne conception du matériel et des locaux et surtout sur une discipline de travail très stricte. L'expérience a montré que, dans le domaine de la sécurité en milieu radioactif, l'éducation du personnel est d'une importance capitale. La bonne marche d'une usine dépendant autant du comportement des hommes que de la qualité des appareillages, il importe au plus haut point que chaque travailleur connaisse parfaitement son travail et soit bien conscient des risques auxquels l'expose l'inobservation des consignes ou des simples règles d'hygiène radioactive.

Les efforts accomplis depuis quelques années à Marcoule pour mettre en application ces principes ont permis d'obtenir des résultats très encourageants. Leur poursuite est le gage certain d'une sécurité toujours accrue.

ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/83 France

Review of safety measures developed during six years' experience of the spent-fuel reprocessing plant at Marcoule

By J. Rodier

In a plant for processing irradiated fuels, as in any other, continuity of production can be achieved only through constant efforts to maintain safety standards. The authors examine the safety factors inherent in the design of the installations from the point of view both of radioactive and of conventional protection.

In the nuclear industry, it is not enough simply to take the usual precautions against the risk of explosion, fire and other disasters; great attention must also be paid, to the problems of containment, radioactivity and criticality risks.

The general principles of the technical prevention of radioactive risks are now thoroughly understood. They must be put into practice at the time when the plant is being designed, by taking appropriate measures relating to:

(*a*) The design and construction of the various installations in the light of the risks mentioned above;

(*b*) The division of the plant into zones;

(*c*) The movement of personnel and materials; and

(*d*) Ventilation, and decontamination of the air.

Technical safety factors, however, are not enough in themselves. However good the material and the degree of automation achieved, the smooth running of a plant depends on the men responsible for supervision and repair work, and for taking prompt action when required. Technical safety is never final; it must be continually sustained. It can never be effective without the wholehearted collaboration of the operatives. Technical precautions must therefore be backed up by psychological measures, based on: education of personnel; appropriate organization of operations and definition of the safety rules; enforcement of obedience to the rules and general supervision of personnel behaviour; and constant reminders of the need for preventing radioactive risks.

Six years of running the irradiated-fuel processing plant at Marcoule have shown the importance of close integration of human and technical factors. The authors analyse the results obtained from an organization of the operations aimed at eliminating the causes of incidents. The following points in particular are studied:

(i) Frequency of entry into forbidden zones during normal operations;

(ii) The distribution of exceptional tasks;

(iii) The prevention of criticality risks;

(iv) Zone contamination incidents;

(v) The volume and nature of decontaminated materials and wastes;

(vi) The course and distribution of doses of external irradiation received by workers;

(vii) The frequency of cases of body contamination; and

(viii) Internal contamination fixed in the workers' organism.

Conclusions are drawn from various incidents that have occurred during the period over which the plant has been in operation.

A/83 Франция

Мероприятия по безопасности, выработанные в результате опыта шестилетней эксплуатации завода по переработке облученного горючего в центре в Маркуле

Ж. Родье

На заводе по переработке облученного горючего, так же как и на любом другом заводе, непрерывность производственного цикла может быть достигнута при условии постоянного соблюдения правил техники безопасности. Авторы рассматривают безопасность проектируемых установок с точки зрения принимаемых в настоящее время мер как обычной, так и радиационной безопасности.

В атомной промышленности, помимо принятия обычных мер предосторожности, направленных на устранение опасности взрыва, пожара и т. д., следует уделять большое внимание проблемам безопасности, возникающим в связи с хранением материалов, радиоактивностью и опасностью возникновения надкритичности.

В настоящее время хорошо известны основные технические меры предосторожности, направленные на устранение опасности облучения. Они должны осуществляться с момента разработки конструкции установок с учетом следующих принципов: i) проектирование и сооружение установок должно проводиться с учетом требований техники безопасности; ii) необходимо разделить завод на зоны; iii) необходимо учитывать передвижение персонала и материалов; iv) необходимо осуществлять вентиляцию и очистку воздуха.

Технические меры сами по себе недостаточны для обеспечения безопасности. Каковы бы ни были качество используемых материалов и степень достигнутой автоматизации, работа завода требует участия людей, которые наблюдают ход процесса, ремонтируют оборудование и проявляют инициативу в том или ином вопросе. Безопасность будет полной в том случае, если персонал, работающий на заводе, будет соответствующим образом подготовлен. Поэтому необходимо, чтобы технические мероприятия по безопасности дополнялись психологическими мерами. Эти психологические меры включают: — обучение персонала; — организацию работы и составление правил безопасности; контроль за соблюдением правил безопасности и за поведением работающих и постоянное напоминание о необходимости соблюдения мер предосторожности против опасности облучения.

Шестилетняя работа завода по переработке облученного горючего в Маркуле показала значение фактора овладения человеком техникой. Авторы анализируют результаты, полученные благодаря организации работы, направленной на то, чтобы избежать аварий. В частности, они исследуют следующие вопросы:

a) число заходов в запрещенную зону во время нормальной работы;

b) распределение работ чрезвычайного характера;

c) меры предосторожности, направленные на устранение надкритичности;

d) случаи заражения зон;

e) объем и характер дезактивированных и удаленных отходов;

f) подсчет и распределение доз внешнего облучения работающих на заводе;

g) частота случаев заражения тела;

h) внутреннее заражение организма работающих.

Эти данные были получены в результате анализа последствий различных аварий, имевших место в течение многолетней работы этого завода.

A/83 Francia

Consideraciones relativas a la seguridad deducidas de la explotación durante seis años de una instalación de tratamiento de combustibles irradiados del Centro de Marcoule

por J. Rodier

En una instalación de tratamiento de combustibles irradiados, como en cualquier otra instalación industrial, la continuidad de la producción no puede alcanzarse sino merced a un estudio constante de la seguridad. Los autores examinan los elementos de seguridad aportados por la concepción de las instalaciones tanto desde el punto de vista de la seguridad clásica como de la seguridad radiológica.

En efecto, en la industria nuclear no basta adoptar solamente las precauciones habituales contra riesgos de explosión, de incendio, etc.; hay que prestar además especial atención a los problemas planteados por el confinamiento de las sustancias, la radiactividad y los riesgos de criticidad.

Los principios generales de la prevención técnica de riesgos de irradiación son en la actualidad bien conocidos. Su aplicación debe traducirse, desde el momento en que se proyectan las instalaciones, en disposiciones relativas a: i) la concepción y la cons-

trucción de las instalaciones, habida cuenta de los riesgos enumerados; ii) la división de la fábrica en zonas; iii) la circulación del personal y del material; iv) la ventilación y la depuración del aire.

Sin embargo, los factores técnicos de seguridad no bastan por sí solos. Sea cual fuere la calidad del material y el grado de automatismo alcanzado, la continuidad de funcionamiento de una fábrica implica la intervención del hombre que vigila, repara, toma iniciativas. La seguridad técnica no es nunca definitiva y debe velarse por ella en cada instante. Sólo puede conservar su eficacia con la colaboración de los trabajadores. Es, pues, menester que la prevención técnica sea completada por una prevención psicológica. Esta prevención psicológica se basa en la formación profesional del personal, la organización de las tareas y el establecimiento de consignas de seguridad, el control de la observancia de las consignas y del comportamiento de los trabajadores, la advertencia constante de la necesidad de prevenir los riesgos de irradiación.

Seis años de explotación de la instalación de tratamiento de combustibles irradiados de Marcoule han demostrado la importancia de una colaboración estrecha entre el hombre y la técnica. Los autores analizan los resultados obtenidos gracias a una organización del trabajo encaminada a eliminar las causas de incidentes. Se estudian en particular los siguientes puntos:

a) La frecuencia de entrada en las zonas prohibidas en circunstancias normales de funcionamiento;

b) La distribución de las tareas de carácter excepcional;

c) La prevención de riesgos de criticidad;

d) Los incidentes de contaminación de zonas;

e) El volumen y la naturaleza del material descontaminado y de los residuos eliminados;

f) La evolución y la distribución de las dosis de irradiación externa recibidas por los trabajadores;

g) La frecuencia de los casos de contaminación corporal;

h) La contaminación interna fijada en el organismo de los trabajadores.

Se deduce una doctrina basada en los diferentes incidentes ocurridos durante los años de explotación de la instalación.

Safety aspects of US fuel reprocessing

By J. A. McBride,* E. B. Sheldon,** W. S. Frank *** and J. C. Bresee ****

In the large-scale chemical processing of irradiated nuclear fuels, the plant operator has a number of safety problems in addition to the usual hazards of fire, explosion, and personnel injury by mechanical and chemical agents. The processing plant must:

(*a*) Handle fissionable material in quantities representing many times the minimum critical mass, but maintain conditions that avoid criticality;

(*b*) Protect personnel from direct exposure to high radiation levels, and from intake of highly toxic radiation emitters;

(*c*) Confine radioactive materials within the processing equipment, and avoid their uncontrolled release to the environment;

(*d*) Store radioactive wastes in a way that avoids their release to the environment for as long as they present a hazard.

Two aspects of the fuel processing problems are perhaps unique. Complicating the plant operator's problem is the fact that radiation and criticality hazards tend to arouse a highly emotional reaction in people which may result, through public pressure, in uneconomical and impractical safety standards. Aiding his cause is the fact the character of the radiation or criticality hazard is known in advance with some degree of precision, and can be insured against by suitable measures.

The prior knowledge of the existence of radiation and criticality hazards has led to the development of well defined and rigorous procedures governing all phases of fuel reprocessing. Operating procedures are developed early in a facility design and construction program, and are revised as necessary throughout the life of the processing plant, as indicated by results of continuing development work on operational problems. Intensive training of the operating personnel begins well in advance of actual operation, and continues throughout the plant life. Procedures are established for continuing surveillance of operations for safety, and administrative follow-up assures that all elements of the over-all program are accomplished effectively. These have provided adequate protection from the hazards. In a typical year, less than one-tenth percent of USAEC-connected workers received a radiation dose at work exceeding that received in a complete dental X-ray. The fact that the atomic energy industry ranks with the top US industries in its over-all safety record attests ot the value of its safety program.

The subject of safety in nuclear fuel processing plants may be divided into three broad categories: radiation safety, nuclear criticality safety, and ordinary industrial safety. A complete discussion of any one of these three would be beyond the scope of this paper; the subject of nuclear criticality safety, for example, was discussed in considerable depth in 1961 at an OECD symposium at Karlsruhe. Subjects of some 23 papers ranged from design philosophies for chemical plants to the administration of operational safety programs. We have in this paper considered only the principal procedures used for control and the results, in terms of both the significant incidents which have occurred and the effect of fuel processing plant operations on the general public in the areas surrounding the plant.

SAFETY PROCEDURES AT US FUEL REPROCESSING SITES

In nuclear fuel reprocessing, as in all other segments of the industry, concentration on and planning for safety begins in the earlier stages of design. Facilities, equipment, and processes are all evaluated continuously in terms of the degree of operational safety required in the plant. Design objectives are to ensure adequate radiation protection of personnel and to provide secondary containment of radioactivity in the event of difficulties with process equipment; processes are designed to provide positive assurance that radioactivity will not be released to the environment in amounts that would be hazardous to important life forms, and equipment is designed to provide positive containment of all radioactive streams and to minimize the hazards of criticality. Critically safe geometry or fixed nuclear poisons are used where practical. Where administrative controls are necessary, instruments are

* Idaho Chemical Processing Plant, Idaho Falls, Idaho.

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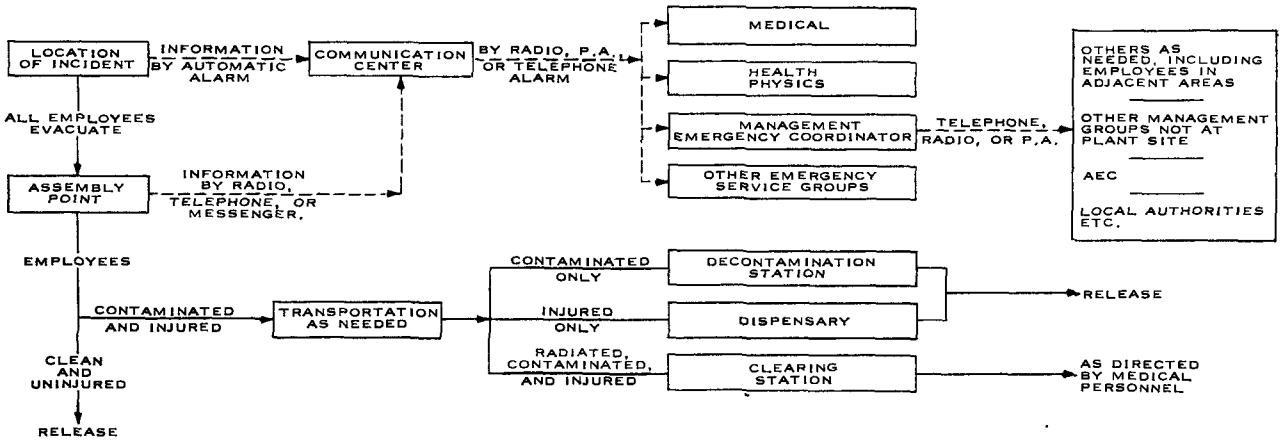


Figure 1. Immediate action plan for radiochemical emergency

used to provide the operator with information as to the status of the process and to minimize the potential for error.

Procedural controls

Specific details of the administrative controls used vary from plant to plant in the US but all systems have these points in common: (*a*) There is a clear definition of the responsibilities of the operating group and of the technical group in respect to safety and plant operation. (*b*) There are provisions for periodic management audits of performarce and operational safety. (*c*) There are provisions for competent and thorough reviews of the safety aspects of proposed new facilities and facility modifications. It is interesting to note several basic differences among the various systems. Technical specifications prepared at the Savannah River Plant (SRP) are quite detailed, covering completely the chemical and physical limits of the process, and require off-site management approval. In contrast, Hanford technical specifications are approved locally and define limits necessary to maintain product quality and nuclear and chemical safety, but do not specify detailed procedures. At Hanford, operation on a waiver of approved specifications is not permitted where criticality or chemical safety is involved; the procedure for obtaining approval of revised specifications is, however, somewhat less complex than at the SRP. The Idaho Chemical Processing Plant (ICPP) procedures are somewhat less formalized than those of either Hanford or SRP, but they follow the same general pattern. ICPP is unique in granting to its safety audit group (the Engineering Safeguards Committee) effective veto power over the operations organization through its assigned responsibility to approve or disapprove operating procedures.

Emergency plans

Increasing emphasis has been placed in recent years on improvement of procedures for handling emergencies. Effective preplanning of procedures can correct operational practices which might create emergencies and can result in a well designed plan of action to restrict the magnitude of an emergency and to minimize its effects. Instruments have been developed which detect immediately a criticality incident, and give an alarm. Intensive training assures that employees in processing operations are thoroughly familiar with all emergency plans and procedures; immediate implementation of personnel evacuation procedures minimizes exposure to radiation.

Following any incident, it is now routine to identify as promptly as possible employees who may have received significant exposures, and immediately to implement prearranged procedures for providing these individuals with competent medical attention, including decontamination if necessary, and counselling. Reactions of several employees who have been involved in recent radiation incidents indicate the particular importance of the latter. It is also quite important to determine as soon as possible the actual exposures received by any employee involved in a radiation incident. A typical immediate action plan for radiochemical emergency is shown in Fig. 1.

Criticality control

Safe processing of fissionable material may utilize concentration control, limitation of batch size, and geometrically favorable * configuration, either singly or in combination. Nuclear poisons also may be used, either as fixed elements of the process equipment or as soluble components of the process streams. Instruments are used to monitor the process for unsafe conditions.

An interesting point of philosophy in the evaluation of criticality risks is the fact that, in the establishment

* As used here, "geometrically favorable" equipment is safe by geometry for all *credible* concentrations of fissionable material and conditions in a particular system, but not necessarily for all *possible* conditions.

of the basic design assumptions, both Hanford and the ICPP employ less restrictive safety factors in heavily shielded operations than in nominally shielded operations because of the reduced severity of consequences of a criticality incident in the former. On the other hand, the SRP does not recognize any increased latitude in evaluation of nuclear criticality risks in shielded facilities.

Evaluation of criticality factors has led in the past 10 years to the development of the processing criticality expert in both the AEC and AEC contractor organizations. This new technical expert is skilled in the nuclear physics computations of the reactor physicist, but at the same time has the engineering background which permits the application of these computational techniques to the design and analysis of chemical process equipment.

Process controls

The specific measures being used for criticality control depend on the nature and amount of material being processed. In the case of irradiated natural uranium, plutonium must be controlled at all steps where it is separate or can accidentally become separate from the uranium. If uranium contains more than natural abundance of ^{235}U it too must be controlled. This, for example, is the case with 20%-enriched research reactor fuels.

Where enriched uranium is processed at ICPP in equipment of favorable geometry, concentrations up to 350 g/l can be tolerated; concentrations in those portions of the plant having unsafe equipment are limited to 6 g/l. At the SRP, existing Purex facilities were converted with minimum capital outlay to the processing of enriched uranium, and concentration control in the geometrically unsafe equipment is achieved primarily by material balance and instrumentation. Special inserts in the dissolvers limit the number of fully enriched fuel elements which can be charged, and hold these fuel elements two feet from the bottom of the dissolver. Steam coils are also two feet from the bottom of the dissolver, leaving a volume beneath sufficiently large to maintain a critically safe concentration level of uranium in the event of inadvertent over-evaporation of the dissolver contents. Positive blinds (referred to as "nuclear blanks") are placed in all lines carrying reagents whose introduction into a process vessel

Table 1. Routine measures for criticality control, natural uranium processing

Process step	Possible hazard	Method of control	
		SRP	Hanford
Dissolving	Negligible	Maintain acidity above 0.2 molar to avoid Pu polymer	Maintain batch charge limit and Pu/U ratio limit
Feed clarification (centrifugation)	Pu polymer in centrifuge	Analyze feed for acidity and inextractable Pu	(Not applicable)
Solvent extraction			
1A (Pu-U extraction-scrubbing)	Pu reflux	(Hazard negligible)	No problem, Pu/U ratio limits
1B (Pu partition)	Pu reflux	Neutron monitor adjacent to contactor	Batch limit maintained by material balance. Acidity maintained above 0.1 *M*, ferrous concentration held below 0.033 *M*
1C (U stripping)		(No problem)	
2A (Pu extraction-scrub)	Pu reflux	Neutron monitor adjacent to contactor Conductivity meter on 2AS to monitor scrub acidity	No problem; geometrically favorable equipment
2B (Pu stripping)	Pu reflux	Neutron monitor Conductivity meter on 2BP	Geometrically favorable equipment
Second plutonium cycle, feed and run tanks	Pu precipitation	Maintain < 200 g Pu/ft² of tank bottom area Blank all lines that may introduce precipitating chemicals	No problem; geometrically favorable equipment
Solvent recovery	Pu precipitation	Analyze stripped solvent for Pu	Negligible problem, batch limit on each vessel by sampling and analysis
Rework	Pu precipitation	Verify acidity in all vessels prior to adding Pu Maintain < 200 g Pu/ft² of tank bottom area	Maintain batch limits on all vessels by sampling and analysis, maintain Pu/U ratio limit

Table 2. Routine measures for criticality control, enriched uranium processing

Process step	Hazard	Method of control	
		SRP *	ICPP
Dissolving	Unsafe batch size of ^{235}U	Control weight of ^{235}U charged Inspect for heel before charging Dissolver configuration	Control weight of ^{235}U charged Heel control on batch dissolvers Continuous dissolvers are of geometrically favorable configuration
Evaporation	Over-evaporation of ^{235}U	Limit evaporator temperature Control weight of ^{235}U in evaporator feed batch	Geometrically favorable equipment, fixed poison in head of continuous evaporator Limit on ^{235}U in batch process
Feed clarification (simmer and centrifugation)	Evaporation during simmer	Limit simmer tank temperature	(Not applicable)
Solvent extraction	^{235}U reflux	Neutron monitors on all contactors Dual flow measurement of all cold feeds U colorimeters on ICU and IEU Conductivity meters for acidity in ICU, IEU Control TBP concentration	Geometrically favorable equipment Control TBP concentration Flow alarms and flow rate checks on process streams Control feed concentration (sometimes)
Solvent recovery	Precipitation of ^{235}U	Na_2CO_3 used for washing solvent (no NaOH) Analyze stripped solvent for U	Use HNO_3, then Na_2CO_3 for washing solvent (no NaOH)
Waste evaporation	Precipitation of ^{235}U or development of unsafe concentrations	Analyze waste for ^{235}U before neutralizing and evaporating	Analyze for U before evaporation
Rework	Precipitation of ^{235}U or development of unsafe concentrations	Control weight of ^{235}U transferred to rework tanks Verify acidity in receiving tank before adding ^{235}U	Fixed poisons in run tanks and in evaporator head Control ^{235}U weight and concentration Control reworked product transfers
All	Precipitation of ^{235}U or inadvertent transfer to unsafe vessels.	Use "nuclear blanks" to prevent addition of unsafe materials to process vessels and cold feed tanks	Use lock and seals on valves and jets where hazardous transfers would be possible. Color code valve handles according to hazard

* Primary control of criticality in all steps is ^{235}U concentration of < 6 g/l.

might result in precipitation of a fissionable material; of particular concern are caustic lines which run to process vessels or lines carrying acidic plutonium solutions.

Tables 1 and 2 compare for natural and enriched uranium processing, respectively, the routine criticality control measures used in plants of different basic process design. Of interest in these tables are not only the different methods used to solve essentially similar problems, but also the influence of equipment design on the character of the problems encountered.

Soluble poisons have not yet been generally accepted as primary safeguards against criticality but have been used as emergency devices or as secondary safeguards. A significant development in this area is the announced plan of the ICPP to employ fluoboric acid as a soluble neutron poison in the modified zirconium alloy head-end system now being installed. Using the most conservative set of assumptions conceivable for the maximum charge of enriched uranium, the calculated k_{eff} for the dissolver is 0.91 when 3.8g/l boron is present in the dissolvent; under most probable operating conditions, the values of k_{eff} should be in the range of 0.70–0.80. Duplicate neutron attenuation instrumentation and two independent administrative controls protect against inadvertent omission of the poison.

Equipment design

In the processing of plutonium after it has been separated from uranium and decontaminated in the canyon process, plutonium concentrations are generally above critically safe values. Recent changes in US fuel processing facilities have been in the direction of improved criticality safety through geometrically favorable equipment. Thus, the latest Hanford solvent extraction plant is geometrically favorable, and the SRP plutonium metal recovery line was recently rebuilt using geometrically favorable slab tanks for intermediate storage of process solutions, and subsequently modified to eliminate points where it was possible to go from critically safe to non-critically safe equipment. Presently undergoing shakedown at Hanford is a new plutonium reclamation facility, in which geometrically favorable equipment has been used throughout. This facility in addition represents a unique combination of a remote-maintenance and direct-maintenance facility, with the most troublesome equipment, from a maintenance standpoint, located for direct servicing and replacement in glove boxes along the outside wall of the remote cell.

Adoption of completely safe geometry is impractical in many cases, but the somewhat larger geometrically favorable vessels (in the form of slabs or small-diameter cylinders) permit much larger batch sizes, giving higher rates than in "safe" equipment with no sacrifice in safety. Interaction of these slabs or cylinders with other vessels, piping, and sumps must be carefully evaluated in establishing safe limits, and the possibility of reflection by walls, water flooding, structural members, and personnel also must be considered.

Fixed poisons have generally been used only as a secondary criticality control, except for the use of Pyrex glass (a borated glass) Raschig rings in storage vessels at the Rocky Flats Plant and at ORNL and the use of boron stainless steel cartridges in evaporator heads and storage vessels at the ICPP. Examples of use as secondary controls include use of Pyrex glass Raschig rings on the floor of a storage cell at ICPP, which has not only improved the nuclear safety of the storage cell, but also has permitted an order of magnitude increase in the total quantity of fissionable material which may be stored in the cell in geometrically safe storage vessels. Cadmium-lead Raschig rings covered with Poly-ethylene are similarly used in sumps under large storage vessels for enriched uranium solution at the SRP.

Instrumentation

The broad subject of process instrumentation in fuel reprocessing is beyond the scope of this paper, but several applications are of direct interest to the problem of criticality safety.

Criticality detection instruments are a relatively recent innovation and are used at both SRP and Hanford, except in heavily shielded high radiation areas. The gamma sensitive instruments are spaced so as to have two sensors "looking" at each suspect area and are set to alarm in a field of 3–5 r/h (equivalent to an approximate 1×10^{15} fission burst). These instruments provide immediate indication of a criticality incident and signal an immediate evacuation of the area by operating personnel.

Neutron monitors also are used at both SRP and Hanford, where they warn of build-up of dangerous concentrations of plutonium and uranium in systems where it is possible for these materials to accumulate. To minimize maintenance of these instruments in shielded areas, SRP inserts easily removable probes through pipe sleeves into the monitored areas; Hanford has developed a monitor which is giving reliable service totally within the high radiation zones. Among other instruments used primarily for safety purposes are high- and low-level flow alarms on those streams where either insufficient or excessive flow might result in a criticality hazard, continuous reading colorimeters to measure uranium concentrations of streams going into geometrically unsafe vessels, and conductivity meters for acid streams where loss of acidity can permit reflux or precipitation of plutonium.

SAFETY EXPERIENCE AT US FUEL REPROCESSING SITES

Confinement of radioactivity

Millions of curies of fission products enter the fuel reprocessing plants every day. These are relatively immobile at the time of receipt, and the major hazards they present are guarded against by suitable shielding during transportation, receiving, and storage. Throughout the rest of the process, however, they are in a highly mobile form, and extreme precautions must be taken to ensure that no stream is released to the environment containing sufficient activity to constitute a hazard to important life forms. More than 99.99% of this fission product activity goes in the high-level waste stream to underground storage tanks at the plant site. The remainder may find its way to the environment by any of several routes: discharge of gases to the air through the plant stack; discharge of liquids to the ground as process condensates, cooling water, or storage basin overflow; or discharge of solids to the ground through burial of such materials as gloves, cleanup rags, analytical laboratory wastes, and contaminated equipment. Typical data for release of radioactivity from irradiated fuel processing plants are summarized in Table 3.

Surrounding each of the fuel processing plants in the US is an extensive network of monitoring stations.

Table 3. Activity releases to environment from irradiated fuel reprocessing plants [a]

(Data given in Curies)

Plant	1963			1962			1961			1960			1959		
	SRP	Hanford	ICPP	SRP	Hanford	ICPP	SRP	Hanford	ICPP	SRP	Hanford	ICPP	SRP	Hanford	ICPP
Airborne, via stack	9	139	102	19	102	258	167	255	326	17	350	262	177	296	11 404 [b]
To ground, via seepage	2.61	37 200	18	227	63 700 [c]	48	183	8 200	18	466	7 800	4	230	8 700	12
Direct disposal to:															
Surface water	1	–	–	1	–	–	3	–	–	8	–	–	1	–	–
Underground water	–	–	6	–	–	29	–	–	30	–	–	32	–	–	46

[a] Source of data: SRP – LTR, B. L. Taber (Dupont, Wilmington) to W. G. Belter, 27 February 1964.
Hanford – HW-SA-3371, G. J. Alkire, 24 February 1964, Memo to W. G. Belter.
ICPP – From monthly radioactive waste reports.

[b] > 90% from burst during October criticality incident.

[c] Increase due to process change, principally ruthenium.

These regularly analyze for radioactivity the surface and underground waters leaving the plant areas, the air, and the vegetation and wild life in the neighborhood of the plant site. At none of the sites has the level of any of the monitored isotopes exceeded a few per cent of the appropriate values in the Radiation Concentration Guide (RCG) for off-site radiation levels. In all cases examined, the contribution of activity from the fuel processing plant to the background radiation levels was a small proportion of the background level, and was in general less than the activity added by fall-out from each series of recent weapons tests. Analyses of locally-produced foodstuffs for the biologically significant isotopes, together with the data on environmental activity, make possible reasonable estimates of the contribution of fuel reprocessing activities to the total exposures received by "average" residents of the surrounding population. Such estimates consistently indicate that the major sources of exposure are background and worldwide fall-out.

A general design criterion of radioactive processing plants is that at least two unrelated abnormal events must occur before activity can be released into an inhabited area. In spite of this precaution, there have been a relatively large number of spills or escapes of radioactivity from the primary and secondary containment areas of processing plants. Most of these, however, have been minor, requiring only routine clean-up. Hanford, for example, in twenty years records only 32 cases in which radioactivity escaped the confines of plant buildings or waste containers. In none of these cases did any plant employee receive an excessive dose of radiation in either the incident or the clean-up nor was there any significant off-site consequence. Major spills which have occurred are summarized in Table 4 (part A).

Spills of radioactive materials in routinely accessible areas of plants are likely to cost more in terms of plant down-time and dollars required for clean-up than in terms of contamination of the environment or radiation exposures of employees. At the ICPP, a leak in a radioactive process line which connected to equipment located outside a cell resulted in severe contamination of the access corridor of the plant. Operation of the plant continued in alternate equipment, but this portion of the plant was out of service for 130 days, and renovation costs were $300 000. In a similar incident at the SRP, cooling water, contaminated via a leak in an evaporator heat exchanger, flowed into the gang-valve corridor at a rate exceeding the drain capacity of the corridor, and eventually found its way to lower levels of the building normally accessible to the plant operators. Although no spread of contamination beyond the confines of the building occurred, here again the incident was costly in terms of clean-up costs. Rupture of an off-gas filter at the ICPP fuel element cutting facility in October 1958 released approximately 100 curies of particulate solid activity to the environment. This material consisted of quite insoluble, heavy particles, which settled out within a short distance from the point of release.

Fires and explosions

Fires and explosions are of particular significance in fuel processing plant safety because of their potential for spreading contamination widely. In fact, reviews of hazards in fuel reprocessing plants have indicated that uncontrolled spread of radioactivity to an extent resulting in a major incident is more likely to be caused by a fire which involves radioactive materials than by a criticality incident.

Table 4 (part B) lists the significant fires and explosions which have occurred in US reprocessing plants. Of particular interest was the ORNL plutonium evaporator explosion in November 1959. Cause of this explosion was determined to be the deflagration of nitrated organic products inadvertently formed from a

decontaminating agent which contained phenol. The explosion resulted in the release of approximately 150 g of plutonium from the evaporator and associated equipment, of which approximately 0.6 g escaped from the building and contaminated the immediate surroundings. No one was injured by the explosion, and no one received more than 2% of a lifetime body burden of plutonium or an overexposure to sources of ionizing radiation either at the time of the incident or during subsequent clean-up operations. Following completion of the analysis of this incident, radioactive safety criteria at ORNL were considerably revised to provide for a guaranteed secondary containment around all radioactive operations, and a revised communication system to aid in identifying and dealing with radioactive emergencies.

A fire in a facility processing fissionable material presents the operator with an uncomfortable dilemma: whether or not to use water in fighting the fire. The usual procedure has been to exclude water from fires involving fissionable materials, but in recent years there has been a growing recognition of the need to re-examine the criteria used for the determination of whether water should be used in fighting such fires. The recent fire in plutonium processing facilities at Hanford, which apparently was started by an exothermic reaction in an anion exchange column, showed that the use of water in such fires may be desirable and may possibly be the lesser of the two potential evils.

Criticality incidents

Five nuclear criticality incidents have occurred in the US during the chemical processing of nuclear fuel materials. One of these occurred at the Oak Ridge Y-12 Plant in operations attendant to the recovery of scrap uranium, two occurred in operations attendant to reclamation of unirradiated plutonium, one at Los Alamos and one at Hanford. Only two have occurred during the processing of irradiated fuels, and both of these were at the ICPP. Table 4 (part C) summarizes these incidents.

Several significant inferences may be drawn from the analyses of these incidents. First, in terms of

Table 4. Recent significant incidents in irradiated fuel reprocessing

Location	Description	Appraisal
	A. Radioactivity spills	
Hanford 1956	In preparing to air-purge evaporator following removal of plug in line using steam, operator inadvertently closed only vent to off-gas system. Pressure vented through manometer in normally accessible area, releasing considerable Pu solution.	Operational error, plus faulty procedure involved. Two vents normally on vessel, one blanked off, operator did not know this, sought to use blocked vent. No radioactivity spread outside building, but contaminated area still controlled as a radiation area, even after clean-up.
ICPP Oct. 1958	Off-gas filter ruptured at fuel element cutting facility, releasing 100 curies of particulate activity. Acid and moisture droplets resulting from cell decontamination weakened filter.	Filter not designed to withstand acid and moisture. Probability of exposure not anticipated.
ICPP Oct. 1958	Access corridor contaminated by solution from pulse line when diaphragm ruptured. Initial small leak absorbed on blotting paper, shielded; increased problem when large quantities released later in attempt to repair line leaks.	Such a problem not anticipated in design of facility. Operation continued after initial trouble because of need for processing service at time. No radioactivity outside building.
ICPP Dec. 1958	Leaking heat exchanger on process evaporator permitted fission products to drain to service waste well with condensate.	No monitor to detect source of leak and differentiate between a number of possible sources going to the waste well. No provision for diversion and treatment.
ORNL Oct. 1959	Leaking heat exchanger on process evaporator permitted fission products to drain to surface waste disposal system with condensate.	Inadequate design; no way to monitor specific equipment capable of contaminating waste system. Careful release of waste from contaminated holding pond insured adequate dilution.
ORNL Nov. 1959	Minor ruthenium activity released from stack following maintenance on blowers. Blowers were decontaminated for maintenance but subsequent work apparently loosened particulates.	Filters inadequate to retain ruthenium. Now scrubbed from streams before release to plant system. Environmental problem a nuisance rather than a hazard.
SRP Aug. 1959	Piping on high activity waste evaporator disconnected using canyon crane shortly after evaporator shut down. Hot liquid spilled, vaporized, and contaminated crane.	Hot canyon seriously contaminated. Low level activity spread through building, none outside. Procedures now prohibit removing covers from cells containing hot equipment to avoid thermal drafts.
SRP Sept. 1960	Contaminated condensate from leaking heat exchanger overflowed canyon corridor to plant floor. $250 000 decontamination expense.	Problem of communication between operators and supervisors. Inadequate planning and procedures for accomplishment of non-routine job. No radioactivity outside building.

Table 4 *(continued)*

Location	Description	Appraisal
	B. Fires and explosions	
SRP Jan. 1953	Evaporator concentrating UNH-nitric acid solution containing Amsco and TBP exploded near end of concentration. Temperature recorder not working and specific gravity recorder off scale (above 2.0).	Hazard not recognized at the time. Subsequent data indicates 135 °C should not be exceeded. No fission product activity involved.
Hanford Feb. 1958	Reactor containing silver nitrate-coated packing used to remove ^{131}I from off-gas exploded, following removal of a plug by a washing procedure using ammonia. $80 000 equipment loss.	Cause unknown. Laboratory unable to duplicate conditions to cause such a violent reaction. No radioactivity outside building.
ORNL Nov. 1959	Evaporator in first solvent-extraction cycle of Thorex pilot plant exploded during cleaning. Decontaminant used contained phenol; heel of this left in dissolver, with possible solvent and TBP decomposition residues, and heated with concentrated nitric acid. 0.6 g Pu spread outside building as result. $350 000 cost of clean-up.	Dangerous procedure resulted from error of supervisor in writing procedural instructions, was further complicated by operator's disregard of an established practice in this operation. Hazardous operations conducted with material of unknown composition.
Hanford April 1960	Dejacketed, partially dissolved slugs left in dissolver only half covered with water. Metal oxidized, caught fire, melted, alloyed with dissolver steel after standing 30 hours. Fire could be controlled only temporarily with water and dilute acid; incident ended only after all metal had been dissolved by repeated acid treatment and water quenches. $250 000 damage and clean-up expense.	Design of new vessel and operating procedures did not recognize possibility of this situation. Inadequate instrumentation to detect and follow early stages of reaction.
Rocky Flats July 1963	Fire resulted from reactions occurring in Dowex IX-4 column loaded with Pu and containing 8*N* HNO_3. Column pressure rose, ejecting contents. High internal temperature indicated by charred resin. Subsequent fire involved combustibles in immediate area.	Exact cause unknown although presumably related to reaction of HNO_3 with loaded resin, or some extraneous organic matter. No procedural faults noted. Lab studies have partially duplicated probable sequence of events.
Hanford Nov. 1963	Ion exchange column pressure rose to 1 000–1 600 psi, apparently as result of some rapid reaction involving ion exchange resin. Gaskets failed, about half of resin ejected from column. Ensuing fire involved combustibles in area; $310 000 loss, mainly clean-up.	Cause unknown, but apparently similar to Rocky Flats incident. $Na_2Cr_2O_7$, inadvertently added to system, may have been factor, but no success yet in reproducing behavior in laboratory. Though fire fought with dry powder, believe water fog would have reduced time to extinguish, damage, and would have created no hazard.
	C. Criticality incidents	
Y-12 June 1958	Enriched U solution permitted to flow unintentionally from another processing area into one of three "safe" tanks; the other two tanks were being leak tested with water. Believing it to be water, operator drained enriched U solution (2.5 kg U) into 55 gallon drum. Unshielded, maximum dose 461 rem, no fatalities.	Overlapping operations, with processing of enriched U one step ahead of area being cleaned up, and separation only by valves and supervisory control. Design, operating procedures, poor communication from one shift to next, human error all involved.
Los Alamos Dec. 1958	During inventory operations, Pu-rich solids were, in violation of procedures, washed into 38 in diam vessel containing aqueous and organic layers. Acid used to wash solids into tank, and air sparging caused caustic-stabilized emulsion to break, and Pu extracted into organic phase. Starting of stirrer changed geometry of organic phase, and Pu (3.17 kg) went critical. Unshielded; maximum dose, 12 000 rem; one fatality.	Inventory non-routine operations; prescribed procedure not followed. Uncorrected deficiencies in design, human error, and inability of supervisory procedures to cope with error were involved.
ICPP Oct. 1959	Over sparging with air to mix material in storage tanks caused pressurizing of infinitely safe tanks, starting siphon to waste tanks where 34 kg ^{235}U at 170 g/l went critical in 5 000 gallon tank Shielded; maximum dose 50 rem β (by air contamination); no injuries.	Design did not provide means to stop flow once siphon started; however, siphon flow was not detected. Uncorrected system deficiencies and poor operating procedures involved.
ICPP Jan. 1961	^{235}U solution (8 kg U at 200 g/l) apparently raised from 5 in diam safe body of evaporator into 24 in diam head by air bubble from line blowing operations to clear a plug. Shielded; no significant radiation exposures.	Unsafe line clearing procedure moved solution into unsafe zone in a presumably safe evaporator. Poor communication between operator and supervisor, and probable procedural error by operator involved.
Hanford April 1962	In course of clean-up and salvage operations in Recuplex unit, product Pu solution (46 liters at 45 g/l) overflowed, undetected, to floor sump and was transferred accidentally by vacuum to an 18 in diam tank through a temporary line to the sump. Unshielded; maximum dose 110 rem; no injuries.	Routing of overflow line not known to operators and not reflected in operating procedures. Mechanical condition of equipment, and failure of supervisory control all involved.

employee exposure, nuclear criticality incidents in the heavily shielded facilities required for processing irradiated fuel are far less hazardous than similar incidents in unshielded or lightly shielded facilities. Second, pre-planning of emergency actions paid excellent dividends in terms of decreased employee exposure (especially in unshielded or lightly shielded plants), proper attention to exposed employees, and management of the situation after the incident. Third, through all of these incidents (and for that matter, through all of the incidents which the authors have studied) there runs a thread of common causes. In nearly all cases, an abnormal operation (i.e., non-routine) was being conducted at the time of the incident; in all of the incidents one or more of the following factors, or a closely related variant, was a cause:

(*a*) Inadequate communication between supervisors on succeeding shifts, or between employees on different elements of the same job, or between staff agencies such as Safeguards Committees, etc., and operating groups.

(*b*) Development of lax attitudes towards safety, as evidenced by a tendency for both supervisors and employees to assume an operation safe until proved otherwise, and by failure to recognize significance of warning signals in the form of slight abnormalities in the process.

(*c*) Inadequate procedures for the job being done. Most of the incidents occurred during abnormal operations, which were not well covered by formal procedures.

(*d*) Unrecognized, or recognized but uncorrected design deficiencies in a process system.

(*e*) Employee error. At least one of the incidents was directly caused by disregard of an established operating procedure.

Actions taken as a result of the findings from these incidents have generally been in the areas of (*a*) improved employee safety education, (*b*) modifications to equipment design philosophy (more widespread use of geometrically favorable design and, in some cases, use of nuclear poisons), (*e*) improved procedures, and (*d*) more rigorous audit of performance.

CONCLUSIONS

Principal conclusions which may be drawn from the study presented herein are:

(*a*) The growth of knowledge and experience in nuclear fuel reprocessing has resulted in greatly improved procedures for safe operation of plants and better process equipment designs for safety. In spite of the large amounts of highly toxic materials handled, and the nature of the hazards involved, the safety record of the fuel reprocessing industry ranks with the top of US industry.

(*b*) In recent years there has been a trend away from administrative controls as a means of avoiding criticality problems in reprocessing; greater use of geometrically favorable equipment and nuclear poisons is noted. Substantially improved instruments aid in administrative control where it is still necessary.

(*c*) In spite of the large potential hazard in the processing of irradiated nuclear fuel there have been relatively few significant incidents, and none has been disastrous. Though criticality incidents attract more attention, fires, explosions, and spills of radioactivity have been far more costly in terms of plant downtime and monetary loss.

(*d*) Analyses of the incidents in reprocessing plants have brought a greater realization of the value of enforcing careful attention to operating procedures, of planning for emergencies, and of training employees in emergency actions.

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Although no specific references are cited in this paper, the following sources have proved especially helpful in acquiring and evaluating both the background and some of the specific information presented herein. Many of these sources present quite complete lists of references.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/278 Etats-Unis d'Amérique

Problèmes de sécurité que pose le traitement du combustible irradié aux Etats-Unis

par J. A. McBride et *al.*

Outre les risques communs aux activités industrielles — incendies, explosions, lésions provoquées par des agents mécaniques et chimiques — le traitement des combustibles nucléaires irradiés comporte deux dangers exceptionnels: d'une part, les résidus sont très toxiques, même en quantités minimes, et d'autre part, le produit principal est fissile. Dans les installations de régénération de combustible des Etats-Unis, on traite couramment des produits radioactifs liquides ou gazeux extrêmement fluides d'une activité de l'ordre de plusieurs millions de curies; jusqu'à maintenant, le nombre des accidents est minime, les doses d'exposition individuelle sont très faibles et la contamination du milieu ambiant est insignifiante. Sous l'angle de la sécurité, la situation aux Etats-Unis est aussi bonne dans l'industrie nucléaire que dans les autres grandes industries. Au cours d'une année caractéristique, moins de $1^0/_{00}$ du personnel travaillant pour l'USAEC a été exposé pendant le travail à une dose d'irradiation supérieure à celle reçue lors d'une radiographie dentaire complète, et personne n'a reçu de dose dangereuse. En période normale, l'usine de traitement chimique de l'Idaho (ICPP) manipule 1 750 000 curies par jour; moins de 2×10^{-3} de cette activité est évacuée dans le milieu, en majeure partie par la cheminée. L'installation de calcination des déchets de l'ICPP offre un très bon exemple de la manière dont on peut confiner les déchets fortement actifs. Les effluents gazeux de cette installation qui, depuis décembre 1963, traite les déchets de combustibles à l'aluminium de l'ICPP par calcination sur lits fluidisés sont constamment maintenus à un niveau d'activité d'environ 0,2 % de celui qui est autorisé par les normes de la RCG.

Ce résultat remarquable est dû à plusieurs facteurs. Les installations sont conçues de manière à assurer une protection efficace du personnel contre les rayonnements et comportent des dispositifs de confinement secondaire de la radioactivité en cas de fonctionnement défectueux du matériel. Les procédés de traitement assurent une décontamination pratiquement totale de tous les effluents avant leur évacuation dans le milieu, et le matériel est étudié de manière à assurer le confinement de ces effluents; lorsque les conditions le permettent, on a recours à des poisons nucléaires ou à une géométrie anti-critique. Des programmes de sécurité assurent une planification préalable efficace des procédés visant à prévenir les accidents, des mesures destinées à en diminuer l'importance et des dispositions à prendre d'urgence pour en réduire les effets au maximum; une formation adéquate permet au personnel qui participe aux opérations de traitement de se familiariser avec les plans et les méthodes; un contrôle administratif assure l'exécution efficace du programme dans son ensemble.

Le petit nombre d'accidents survenus permet d'apprécier toute l'efficacité de ces mesures. Ces accidents se sont produits dans des situations anormales ou exceptionnelles, et leurs effets ont été circonscrits à la zone du sinistre. Cinq accidents de criticité nucléaire se sont produits dans les dernières années. Deux de ces accidents ont eu lieu dans une installation de traitement de combustibles irradiés; ils n'ont pas entraîné de dommage matériel ni d'exposition à des doses importantes de rayonnements. Les trois autres se sont produites au cours du traitement de combustibles nucléaires non irradiés et ont entraîné un accident mortel et un certain nombre d'expositions à des doses intenses de rayonnements. Les accidents non nucléaires qui se sont produits dans des installations de traitement chimique ont été des incendies, des explosions chimiques et des fuites de produits radioactifs dans la zone de l'installation. Bien qu'on en parle moins que des accidents de criticité, ils prouvent qu'il faut continuer d'accorder une grande attention à la sûreté des opérations chimiques ordinaires. L'étude de ces accidents a révélé divers problèmes: communications, conduite des opérations par les opérateurs et formation en matière de sécurité; elle a montré dans quels secteurs il est nécessaire d'améliorer la sûreté de fonctionnement du matériel. Elle a notamment permis de mettre au point des dispositifs plus efficaces de contrôle administratif de la sûreté des installations de traitement. Les systèmes varient dans leurs détails d'une installation à l'autre mais tous exigent: *a)* une délimitation précise des responsabilités en matière de sécurité entre techniciens et opérateurs; *b)* des contrôles périodiques des opérations par la direction; et *c)* une étude technique approfondie des problèmes de sûreté que pose toute nouvelle installation et toute modification des installations existantes.

A/278 США

Вопросы безопасности при переработке облученного горючего

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Помимо опасности пожара, взрыва и личного увечья механическим или химическим путем, существующей на любом промышленном предприятии, работа с облученным ядерным топливом создает еще две опасности. Первая из них связана с тем, что отходы производства очень ядовиты даже в незначительном количестве, а вторая — с тем, что главный продукт производства является делящимся веществом. Общие достижения США в деле повышения безопасности работ по регенерации ядерного горючего весьма значительны, несмотря на то, что радиоактивность переработанного топлива исчисляется миллионами кюри, а создается она радиоактивными материалами, находящимися в очень подвижном жидком или газообразном состоянии. Облучение персонала и загрязнение окружающей среды было очень низким. В общем с точки зрения безопасности американская атомная промышленность стоит на одном уровне с любой другой отраслью промышленности в стране. В среднем за год меньше 0,1% всех работников, связанных с Комиссией по атомной энергии США, подверглись облучению, превышающему ту дозу, которую получает пациент при снятии полной серии рентгеновских снимков зубов. Ни у кого из этих работников не наблюдалось каких-либо серьезных последствий. Типичным является процесс на заводе фирмы «Филлипс» в штате Айдахо, во время которого обрабатывается материал активностью 1,75 *Мкюри* в день, и меньше 0,002% этой активности выходит в окружающую среду главным образом через дымовую трубу. Выдающимся примером удержания высокой активности является установка по обжигу отходов вышеупомянутого завода фирмы «Филлипс». Эта установка с декабря 1963 года обрабатывает отходы алюминиевого топлива с применением псевдоожиженного слоя. Дымовой газ этого процесса имеет активность, составляющую около 0,2% допускаемой законом.

Несколько факторов способствуют такому успеху. Установка спроектирована таким образом, что всегда обеспечивается необходимая защита обслуживающего персонала, а в случае неисправности оборудования — удержание радиоактивности. С другой стороны, сами процессы разработаны таким образом, что все потоки очищаются прежде, чем выходят в окружающую среду; оборудование спроектировано так, что предотвращается возможность распространения радиоактивных потоков; там, где это необходимо, используются безопасная с точки

зрения возникновения надкритичности геометрия и поглотители нейтронов. Действующие программы безопасности гарантируют эффективное планирование мероприятий по предотвращению аварий, ограничению ущерба от этих аварий, принятие срочных мер для устранения последствий аварий, полное ознакомление всего персонала со всеми процессами на заводе. Администрация, со своей стороны, следит за выполнением этой программы и этим обеспечивает ее успех.

Достигнутые успехи дают возможность судить, насколько эффективны принятые меры. Несколько аварий, которые произошли при ненормальных или необычных обстоятельствах, ограничивались только тем местом, где они произошли. За последние годы было пять серьезных инцидентов, связанных с возникновением надкритичности. Два из них были в установках, где происходит обработка топлива. Произошли они без повреждения установок и при незначительном облучении работников. Три произошли во время обработки необлученного топлива. В результате их был один смертельный случай и произошло значительное облучение. На заводах химической обработки неядерные аварии включают пожары, химические взрывы и переброску радиоактивности в окрестности завода. Хотя про такие аварии пишут меньше, чем про ядерные, они все-таки указывают на необходимость соблюдать все меры безопасности, установленные для обычных химических заводов. Анализ происшедших аварий указывает на существование недостатков в обеспечении средствами связи, в обращении с машинами, в обучении технике безопасности и определяет возможные пути их устранения. Одним из результатов этого анализа была разработка более надежного механизма контроля со стороны администрации за безопасностью на обрабатывающих установках. Детали такой системы могут быть различны на разных предприятиях, но все они требуют: *а*) точного определения обязанностей и ответственности за безопасность в технических и производственных группах; *b*) проведения периодической административной проверки выполнения всех требований техники безопасности; *c*) компетентного и полного осмотра всего нового или перестроенного оборудования с точки зрения безопасности.

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Aspectos relativos a la seguridad de los métodos norteamericanos de tratamiento de combustibles

por J. A. McBride *et al.*

El tratamiento del combustible nuclear irradiado presenta, además de los riesgos industriales usuales de incendio, explosión y de los daños a las personas por accidentes de tipos mecánico o químico, dos riesgos peculiares: los desechos son muy tóxicos en cantidades ínfimas, y el producto principal es fisible. En los EE.UU. se manejan con normalidad en las operaciones de tratamiento millones de curios de radiactividad en estado líquido o gaseoso muy volátiles; el historial general de seguridad ha sido excelente, las dosis de irradiación sufridas por el personal han sido muy bajas y la contaminación de la biosfera ha sido insignificante. Con su historial general de seguridad, la industria de la energía atómica compite en los EE.UU. con las primeras industrias del país; en un año normal, menos del 0,1 % de los trabajadores relacionados con la USAEC recibieron en el trabajo dosis superiores a las que se reciben en una radiografía dental, y ninguno de los que superaron este nivel recibieron dosis importantes. En una operación representativa del proceso, en la instalación de tratamiento químico de Idaho, se manejan al día 1,75 millones de curios de actividad y, de ella, menos del 2×10^{-3} % se escapa a la biosfera, la mayor parte a través de la chimenea. En la instalación de calcinación de residuos de la Instalación de Tratamiento Químico de Idaho se encuentra una muestra significativa de la contención de partículas muy radiactivas. Los efluentes gaseosos de esta instalación, en la que desde diciembre de 1963 se han venido calcinando en lechos fluidizados los residuos del tratamiento de aleaciones combustibles de aluminio, han tenido siempre una actividad de aproximadamente el 0,2 % de la permitida por los límites de la RCG.

Varios factores contribuyen a conseguir este éxito. Las instalaciones se diseñan asegurando la adecuada protección del personal a las radiaciones y añadiendo una segunda barrera de contención de la radiactividad para el caso de averías en el equipo. Se proyectan procesos auxiliares que prácticamente descontaminen todos los desechos antes de salir a la biosfera y se diseña el equipo para que ofrezca una contención positiva de los circuitos radiactivos; cuando es factible, se utilizan formas geométricas a prueba de criticidad o venenos nucleares. Los programas de seguridad en uso exigen el planeamiento previo y efectivo de procedimientos para evitar accidentes, de las acciones a tomar para reducir la magnitud de un accidente, y de las decisiones a tomar inmediatamente después de un accidente para reducir sus efectos; la capacitación del personal asegura que los empleados de la instalación conocen bien la misma y sus procedimientos; la supervisión administrativa garantiza que el programa general se cumple con efectividad.

El historial de seguridad demuestra la eficacia de estos esquemas. Los pocos accidentes ocurridos tuvieron lugar durante situaciones anormales o no rutinarias y los efectos no salieron de la zona del accidente. En los últimos años han tenido lugar cinco accidentes por criticidad. Dos de ellos, en una instalación de

tratamiento de combustibles irradiados, ni dañaron la instalación ni el personal sufrió daños importantes por radiación; los otros tres tuvieron lugar durante el tratamiento de combustibles no irradiados, en ellos se produjo una muerte y algunas personas resultaron seriamente irradiadas. Entre los accidentes de origen distinto al nuclear que se han producido en dichas instalaciones se cuentan incendios, explosiones químicas y dispersión de radiactividad dentro de la instalación. Aunque se haya hablado menos de estos accidentes que de los de criticidad, su existencia indica la necesidad de continuar concediendo importancia a la seguridad de los procesos químicos normales. Al analizar estos accidentes se han descubierto problemas de comunicación, conducta del operador, y de entrenamiento en seguridad y han salido a la luz algunos aspectos del equipo de la instalación que necesitaban mejoras en cuanto a la seguridad de las operaciones. Uno de los resultados ha sido el desarrollo de mecanismos más positivos para controlar la seguridad administrativa en las instalaciones de tratamiento. Los detalles específicos de los sistemas son distintos para cada instalación, pero todos ellos requieren: *a*) una clara definición de las responsabilidades en materia de seguridad entre los técnicos y los operadores, *b*) informes periódicos de la dirección sobre la marcha de los procedimientos y *c*) la revisión competente y total de los aspectos de seguridad de las instalaciones en proyecto y de los cambios a efectuar en las antiguas.

Safety aspects of the EBR-II reprocessing facility

By M. Levenson,* C. E. Stevenson ** and J. C. Hesson *

The Experimental Breeder Reactor No. Two (EBR-II) project, the construction of which was recently completed by Argonne National Laboratory at the National Reactor Testing Station, consists of a Sodium Cooled Fast Reactor and a Fuel Cycle Facility. The Fuel Cycle Facility contains an argon atmosphere shielded cell where fuel processing and fuel fabrication are done, a connected air atmosphere shielded cell where reactor subassemblies are dismantled and reassembled, and a personnel operating area which surrounds the two cells. The facility is shown on Figs. 1, 2 and 3. The Fuel Cycle Facility will be used to study pyrometallurgical methods of processing irradiated fuels and blanket materials, as well as to study remote methods of fuel refabrication. The over-all project will thus demonstrate the practicability of various proposed fuel cycles. Irradiated fuel will be processed, refabricated, and immediately returned to the reactor so that problems arising from fuel recycle, higher isotope build-up, heavy element build-up, etc., can be evaluated. Since the study of a number of fuel cycles during the life of the facilities is proposed, an attempt was made to design and build the Fuel Cycle Facility sufficiently flexible to be operated safely not only on the first fuel cycle tested, but also on fuel cycles for different fissionable materials, different fertile materials, different fuel alloys or compounds, and different types of fuel subassemblies. With this objective in mind, flexibility and safety considerations influenced all areas of design. The basic Fuel Cycle Facility design was presented at the 1958 Geneva Conference [1] and subsequently described elsewhere in the literature [2, 3]. This paper will be limited to safety questions which arose during the design and construction of the facility.

SHIELDING AND RADIATION MONITORING FOR PERSONNEL PROTECTION

The EBR-II is a fast reactor of high specific power. At design conditions for the first core it is expected that the burn-up will reach approximately 20 000 MWd/ton of total fuel alloy in 135 days of irradiation. Since conventional cooling periods of 100 to 150 days would lead to large out-of-pile inventories, fuel will be processed after a minimum cooling period of 15 days. As a result, general in-cell radiation levels will approach 10^6 roentgens per hour and local levels will be even higher. The basic shielding provided between all process equipment and all operating areas consists of 5 feet of high density concrete (220 lb/ft^3; specific gravity, 3.5). The viewing windows [4] provide a 57 in thickness of 3.2 density glass shielding. All ducts, transfer devices, and cell penetrations are so designed and shielded that at full design levels (10^6 r/h in-cell) the radiation level in all working areas is less than 0.1 mr/h. The adequacy of the design and the integrity of the structure were checked by radiography with a sodium-24 gamma source. All cell penetrations, windows and ducts were checked, as well as all bulk shielding areas. The basic structure provides more than adequate protection for all personnel as long as all radioactive materials remain within the cells. The function of the radiation monitoring system is to provide a record that this is indeed the case, and to raise the alarm if it is not the case. All occupied areas of the building are within line-of-sight surveillance by gamma background monitors. These monitor measure the background radiation level and the results are all continually recorded in the operations control room. If the gamma level exceeds 1 mr/h in any area, alarms are sounded in the area affected and in the operations control room. Moving-tape constant air monitors are used to check on airborne beta-gamma contamination. These results are recorded at each unit as well as in the operations control room so that trends while yet well below tolerance levels are readily observable by the people affected. In addition to the constant monitoring of all in-facility working areas, constant monitors check all streams being discharged to the environment. The high efficiency filters used to remove particulate matter from all exhaust air from the facility, as well as the air itself, are monitored and the results are continually recorded. All exhaust air entering the vent stack is continually sampled and analysed by a group of single channel analysers. The sample is continually filtered and analyses are performed on the particulate material retained on the filter as well as on the raw sample and

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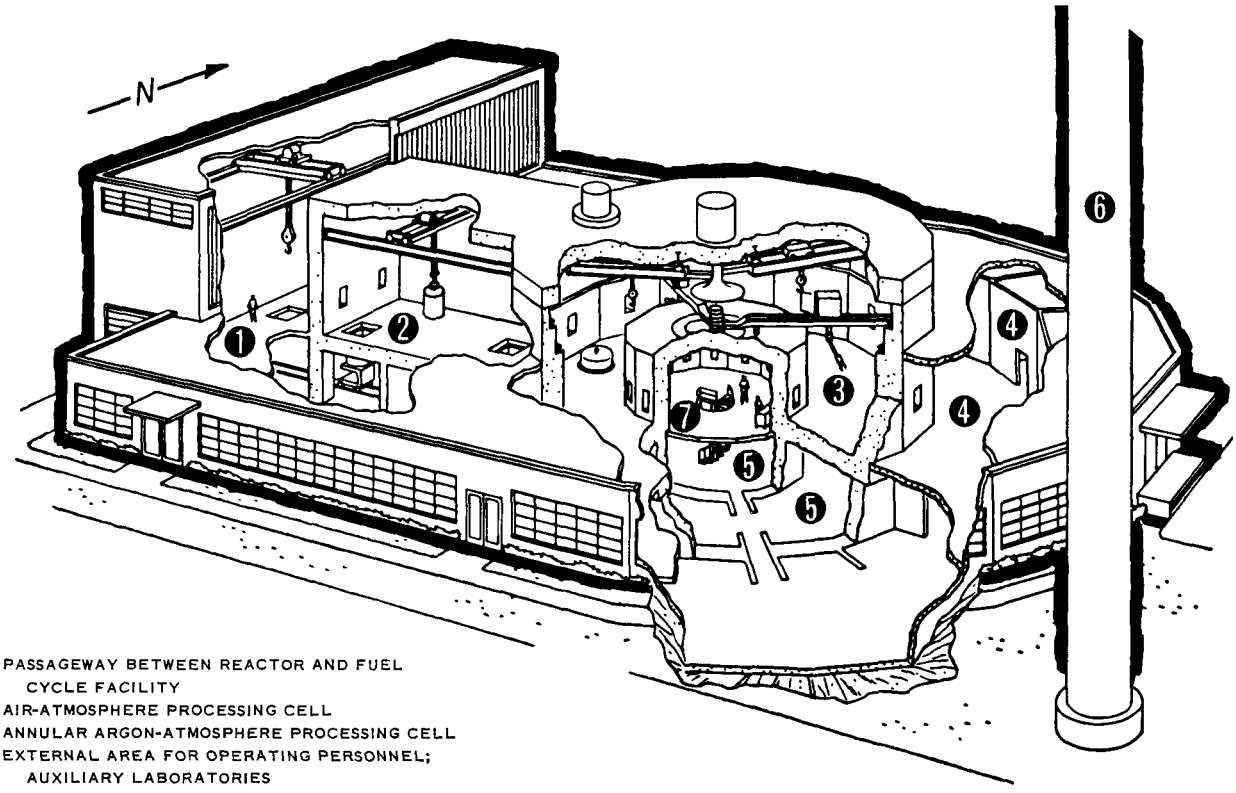


Figure 1. Cutaway view of EBR-II fuel cycle processing facility

on the filtered gas. All results are recorded in the operations control room. In addition, the process off-gas (streams originating at places such as the melt refining furnaces) are monitored for krypton or xenon and iodine activity by single channel analysers. All liquid wastes being discharged to the environment, including sanitary wastes, are also monitored. In summary, all working areas and all effluent streams are under constant surveillance to ensure the safety of all personnel both inside and outside of the facility.

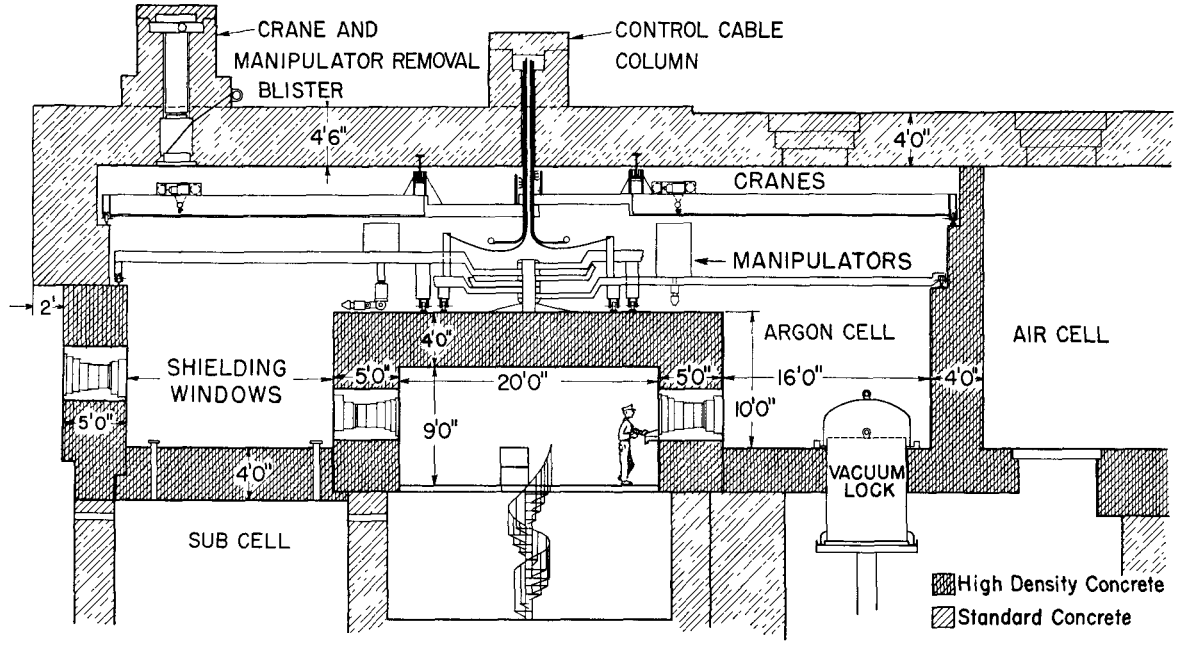


Figure 2. Sectional elevation view of argon cell

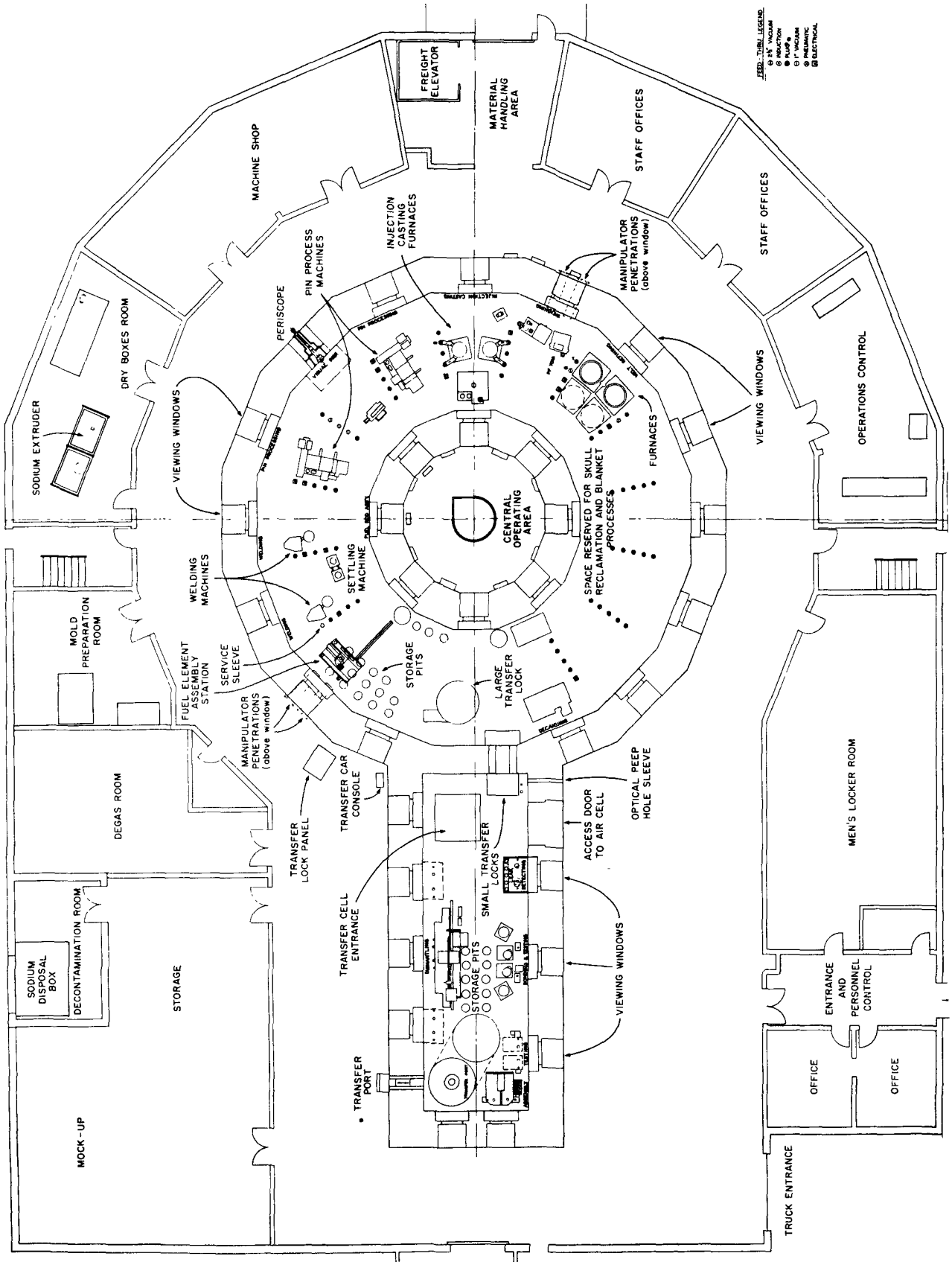


Figure 3. Location of fuel-processing equipment

RADIOACTIVITY AND CONTAMINATION CONTROL

Except for the steps involving the removal of sodium from a discharged fuel assembly, the analysis of samples and the transportation of wastes for burial, the directly connected arrangement of two process cells confines all transfers of and work with radioactive materials to the two well-shielded cells, and thus there is very little opportunity for or hazard from exposure to radiation or contamination during process operations. Samples and wastes (except gases) are all removed from the cell in solid form in sealed containers.

Normally each fuel sub-assembly entering the Fuel Cycle Facility from the reactor contains in excess of 2.6×10^5 curies of fission products. Fuel subassemblies returned to the reactor contain approximately 6×10^4 c. The difference, or approximately 2×10^5 c, must be disposed of in a controlled manner. The noncondensible fission product gases, krypton, xenon, and tritium, are collected from the melt refining furnace and discharged via an off-gas delay tank to the site vent stack which is 200 ft high. The delay tank contains activated charcoal and is baffled to ensure that no short-circuiting occurs. Average residence time in this tank for any batch of gas may be as much as 30 days. Discharge from the delay tank is done only under favorable meteorological conditions. Wind speeds and directions and vertical air temperatures are measured at several elevations and are continually recorded in the operations control room to provide the data necessary for evaluation of atmospheric dilution conditions at any time. The noble metal fission products (Mo, Nb, Ru, Rh, Pd, etc.) remain with the processed fuel and are returned to the reactor. The iodine (approximately 6 000 c per subassembly) is primarily collected in the melt refining furnace fume trap, a disposable device molded of aluminium silicate fibers. After suitable decay times, these traps are canned and buried. Because of the large quantities of iodine involved, several back-up devices are provided for the fume trap. The furnace off-gas, after passing through the fume trap, is passed through an in-cell activated charcoal bed. The gas then leaves the cell and after going through a vacuum pump and aluminium oxide filter, is passed into the off-gas delay tank. In addition to the volume hold-up or delay time for decay, the gas passes through 6 ft of activated charcoal for absorption and retention to increase the decay time. From the delay tank the gas is pumped through a high temperature silver tower (silver deposited on ceramic packing) and then through final high efficiency filters to the stack. All other fission products (rare earths, alkaline earth metals, etc.) are collected as solids for ultimate disposal by burial. Some of the solids collected have such high concentrations of active fission products that heat removal would be difficult, if not impossible, so in such cases the fission product concentrate will be diluted for disposal. Dilution will in general be done with waste metal from other processing or fabrication steps. Only two sources of contaminated liquid wastes exist in this facility. The first is the caustic solution generated by the reaction of residual coolant sodium (on the outside of fuel subassemblies) with steam and water, and the second is the waste generated by the analytical laboratories. Both streams are concentrated by evaporation and the still bottoms are converted to solids for burial. Some of the analytical wastes are sorbed on vermiculite or similar material prior to burial. The pyrometallurgical Fuel Cycle Facility thus produces only solid and gaseous waste streams and no liquid waste storage tanks have been provided.

FIRE PREVENTION

Ordinary fire hazards exist, of course, in essentially all processing and manufacturing operations and facilities. However, it is not to such standard fire hazards but to specialized ones to which we refer in this paper. Most uranium and plutonium alloys are highly flammable, even when unirradiated, and when highly irradiated, tend to be pyrophoric. The sodium metal used as a thermal bond between fuel and cladding materials and the molten metals (magnesium, zinc, cadmium, etc.) used in various process operations and studies are all readily combustible. As a result, it was decided very early in the design of the facility to conduct in an air atmosphere cell only those operations involving clad fuel such as subassembly assembly, testing, etc., and to utilize an inert atmosphere cell for all other operations so that individual equipment items not only would not require individual high integrity inert atmosphere enclosures, but in many cases would require no local enclosures at all. Argon gas was selected as the inert atmosphere in preference to helium on the basis of two factors: the design of purification and cooling systems was simpler, and it was desirable to retain the ability to helium leak test in the cell atmosphere. The cell purification system [5] removes particulate matter, oxygen, and water vapor from the argon and is capable of maintaining an atmosphere that contains less than 10 ppm of water vapor plus oxygen. The argon cell structure is lined with steel plate (½ to 1 in thick). This liner is penetrated by 23 window cavities, four cooling ducts (each approximately 2 by 3 ft), an 8 ft square hatch, a 6 ft diam transfer lock, two smaller transfer locks, over 100 utility electrical cables, 65 major process feed-throughs containing hundreds of pneumatic lines, electrical leads and instrument leads, and dozens of other penetrations ranging in size from 4 to 24 in diam. The toal air in-leakage rate for the entire system (including the cell cooling and purification systems which are actually external to

the cell but directly connected thereto) is less than 0.01 cu ft /min when the cell is operated at 4 to 5 in of water pressure below atmospheric pressure. The resulting argon atmosphere is extremely pure, which not only eliminates all fire hazards within the cell, but eliminates many other problems such as sodium oxide films, partial oxidation of metal while welding, uranium oxide surface skins on metal fuel pins, etc. However, the "safe" condition of many items which prevails while they are in the inert atmosphere cell is radically changed when they are removed from such environment. With this in mind, no facilities exist for transferring anything directly out from the argon cell into normally occupied areas, not even into shields. All transfers to or from the argon cell are done via the adjoining air cell so that if an equipment item contaminated with pyrophoric powder were inadvertently transferred out of the argon cell, the resulting fire would occur in the properly shielded and ventilated air cell rather than in a working area. In most cases, such incidents would not even occur in the air cell proper, but rather in the transfer locks when they are vented to air. While any fire is highly undesirable, the hazards are minimized if one can assure that any fires which do occur take place in a properly ventilated and shielded area, rather than at random locations or in working areas. One potentially serious fire hazard exists outside the argon cell. It involves the irradiated fuel subassemblies which are transferred from the reactor to an interbuilding coffin under a blanket of argon gas. At the time these subassemblies reach the Fuel Cycle Facility, they are still wet with reactor coolant sodium, and are self-heating due to fission product decay. (They may have been out of the reactor core a total of only fifteen days.) Before they are removed from the coffin, the sodium is reacted with air and the resulting sodium oxide is reacted with water vapor and then with water to ensure complete reaction and removal. This operation, which is done in the coffin, is followed by air drying and the subassemblies can then be safely handled in air.

NUCLEAR SAFETY

While all safety problems require serious attention because of the high cost of accidents, fires, contamination, etc., perhaps the most serious potential accident in a fuel reprocessing or fuel fabrication facility is a nuclear or criticality accident.

The prevention of inadvertent criticality in aqueous systems is difficult and expensive when large amounts of enriched uranium or plutonium bearing materials are being processed. In the non-aqueous, non-moderatting media used in pyrometallurgical processes, inadvertent criticality is only a nominal problem, because of the relatively large amount of material necessary to produce a critical mass. The philosophy of design used in the area of criticality control consisted of three basic controls, and the premise that the maximum quantity of fuel to be handled as a unit would have a normal factor safety of at least two even when fully reflected, thus ensuring adequate safety in the event of double batching with a full reflector present. The first and primary control was that normal process materials would be handled only in containers designed specifically for the particular job, and that such containers would be located only on machines or racks which would accept such containers. The second control was that containers and storage facilities for metal scrap, oxides and slags, off-specification material, etc., would always be geometrically safe. Thirdly, that the design of the facility, its auxiliaries and utilities, be such that the introduction of unexpected and significant quantities of moderator into the cell would be so unlikely as to be considered incredible.

The fuel cycle criticality control [6] starts at the interbuilding coffin used to transport subassemblies to and from the reactor. The coffin will hold only one subassembly (6 kg of 46% enriched uranium fuel) and is safe even when filled with water. When a subassembly is transferred into the air cell of the Fuel Cycle Facility, because of self-heating it must be handled with a cooling grapple and can be placed only on machines or in storage racks with cooling provisions. If this is not done, fuel element melting might occur within the subassembly. This would be such a serious consequence that casual storage that might lead to inadvertent criticality is not considered credible. The various machines that will receive subassemblies such as the dismantler, the final assembly machine or the tensile testing machine, will hold only a single subassembly and are thus safe. The storage facilities for subassemblies in the air cell consist of cooled pits in the cell floor. Each pit is outfitted to receive a maximum of four subassemblies (approximately 25 kg of fuel). The pits are nuclearly isolated from each other by concrete and earth fill and have bottom openings connected to the main building air exhaust system in the basement, so that they cannot be filled with water. The primary cooling consists of air exhausted from the pit bottoms so that any partial plugging of the pit drains over a period of time would be detected and corrected. The back-up cooling system is a thin water jacket around each pit. This water cooling system operates under a partial vacuum. The water supply is an open tank below the storage pit level and water is sucked through the cooling jacket. If a jacket or pipe in this system should develop a leak, air will be sucked into the system rather than water being expelled. This could lead to fuel overheating, but not to a criticality incident. At the dismantler, the fuel elements (fuel pins with cladding) are removed from the subassembly and placed in special magazine carts, each holding less than 30 fuel elements. The cart size is such that any number of carts can be located in any area without accumulating a

critical mass. These magazine carts are then transferred into the argon cell via the small air locks. In the argon cell, the magazines are again safely stored and loaded on the decanner machine one at a time. The decanned and chopped fuel alloy pins are collected in the fuel chargers which are permitted to contain only 3 to 5 kg of fuel alloy. Normally, only three of these are in use, but since the dry but fully reflected critical mass of the fuel is over 70 kg, a greater number of and/or larger chargers can be used if it is found to be necessary. The chargers themselves are closed flat trays for purposes of heat dissipation and cannot be accumulated in an unsafe geometry.

The first melt refining operations are being done with 10 kg batches, not for nuclear safety reasons, but for convenience. Even at this small scale (10 kg), only three batches a week need be processed to keep up with the 62 MW reactor. At the present enrichment (46% ^{235}U), batches of 50 kg would still constitute safe charges. The furnaces used are radiation cooled as are all power leads and induction coils, so no water is present in the cells. The product of the melt refining operation is a 10 kg ingot contained in a mold. When removed from the mold it is transferred to an ingot storage container. Either the gravid molds or the ingot storage containers may be placed on the sampling station, on the charge table, or on wall mounted storage racks. Unlike the previous containers, an uncontrolled accumulation of ingot storage containers or gravid molds could produce a critical mass. Therefore, administrative control is used to ensure that all magazines, containers, molds, etc., are placed only on designated machines or in designated storage facilities. No fuel handling container, whether empty or full, can just be placed on the floor. All storage racks have frames so that they will readily accept only one type of item—crucible, fume trap, mold, etc. This same philosophy is followed throughout the fabrication procedure. Fuel pins and elements are handled in always safe magazines through casting, inspection, canning, welding, leak testing, bonding, and bond inspection. At the final assembly machine which is in the air cell, only sufficient fuel elements are available to assemble one subassembly and the machine can accommodate only one subassembly. The finished subassembly is still self-heating enough so that forced cooling is necessary and it is treated in the same manner as a spent subassembly; that is, it is either transferred to the coffin at this time or placed in storage in one of the cooled storage pits in the air cell. The basis fuel units being handled are 10 kg of raw fuel material, 6 kg of fuel in subassemblies, and 1 to 3 kg fuel loads in various magazines. This then requires that from seven to seventy units be mishandled in order to approach a possible critical mass. It should also be pointed out that this approaches the entire inventory normally present in the facility and is therefore considered incredible.

The second condition—safe storage—has been covered to some extent in the preceding paragraph. Storage pit racks are provided for the safe storage of fuel subassemblies, ingots, cans of scrap, etc. The basic design standard for the present fuel (46% ^{235}U) is that batches of metallic ingots will not exceed 32 kg of fuel and that fuel units will be at least 24 in apart center to center with 8 in of intervening clear space. Maximum batch sizes for fuel oxides and small pieces are 10 kg. This still allows for inadvertent double batching and full reflection without achieving a critical mass.

The third design factor was facility design such that non-moderated conditions could be safely assumed. The EBR-II Fuel Cycle Facility has been described but the particular features pertinent to this point will be summarized here.

As previously mentioned, all storage pit water cooling systems in both the air and air cells are negative pressure systems and, in addition, the air cell pits can drain into an essentially infinite volume. No in-cell equipment is liquid cooled. All high frequency cables are of the coaxial type and are radiation cooled. All induction coils are radiation cooled. Welding machines utilize rotating metallic chill-blocks and bonding furnaces utilize gas cooling, as do the assembly and disassembly machines. Heat removal from the various gas streams, cell gas, purification system, etc., is in all cases done with freon 12, which is a gas at ambient temperature, so that even a coil leak cannot introduce a liquid into the cell. The coils are continually monitored by freon leak detectors and if a leak should develop, the coil would immediately be isolated by gas line valves. Fuel storage in the argon cell storage pits is safe, even with water flooding of the pits. Although these pits do not have drains, flooding is very unlikely.

In summary, the EBR-II Fuel Cycle Facility has been designed and built to test non-aqueous processes and remote fuel fabrication techniques and thus handles large quantities of fissionable material, molten metals, pyrophoric materials, and self-heating materials. Attention to design details as well as to design concepts has provided a facility with the desired flexibility and utility without compromising safety.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/280 Etats-Unis d'Amérique

Problèmes de sécurité dans l'installation de traitement EBR-II

par M. Levenson *et al.*

On utilisera l'installation de traitement du combustible EBR-II pour étudier les méthodes pyrométallurgiques de traitement du combustible irradié et les méthodes de retransformation du combustible par télécommande. Le traitement du combustible irradié et la transformation des matériaux fissiles posent toujours d'importants problèmes de sécurité. Un grand nombre de ces problèmes sont beaucoup plus complexes lorsque la matière traitée est un combustible à taux de combustion élevé qui provient d'un réacteur à puissance spécifique élevée et n'a été désactivée que pendant 15 jours. D'autre part, les ensembles compacts non ralentis que la pyrométallurgie permet d'obtenir facilitent la solution de certains problèmes. Les problèmes fondamentaux de sûreté de l'installation EBR-II se répartissent entre quatre catégories: protections et contrôle dosimétrique, mesures à prendre contre la radioactivité et la contamination, prévention des incendies, sûreté nucléaire.

Les protections de l'installation EBR-II ont été conçues pour réduire le débit de dose à moins de 0,1 mR/h. Le débit de dose dans la cellule sera de 10^6 R/h. Après achèvement de la construction, on a radiographié l'ensemble de la structure de la cellule avec une source gamma de sodium 24 et avec une source Sb-Be de neutrons pour vérifier si la cellule a été bien conçue. On procède à un contrôle permanent de toutes les zones de travail de l'usine pour mesurer le bruit de fond gamma et la radioactivité dans l'air, ainsi qu'à un contrôle permanent de toutes les sorties d'air de l'usine pour mesurer l'activité totale, l'activité des particules, celle de l'iode et celle du sodium 24, et des gaz dégagés pour mesurer l'activité du krypton et du xénon. Tous les résultats de ce contrôle sont automatiquement enregistrés et tout phénomène anormal provoque le déclenchement de signaux d'alarme.

Les produits de fission entrant dans l'installation se répartissent entre trois courants. Le premier, qui contient du krypton, du xénon et du tritium, est rejeté dans l'atmosphère et on procède à un contrôle dosimétrique de façon que les limites maximales admissibles ne soient dépassées en aucun point à l'intérieur ou à l'extérieur de l'usine. Le deuxième courant, contenant la masse des produits de fission qui sont des métaux nobles, est laissé intentionnellement avec le combustible (avec lequel ces métaux forment un alliage) et est renvoyé au réacteur. Le troisième (constitué en fait de plusieurs solides obtenus directement) contient l'iode, les terres rares et les métaux alcalino-terreux; on l'isole pour que la décroissance puisse s'opérer et on l'élimine définitivement (sous forme de solides). Les seuls déchets liquides aqueux régulièrement obtenus qui contiennent des produits de fission sont ceux qui proviennent des laboratoires d'analyses.

Les combustibles métalliques irradiés jusqu'à 20 000 MWj/t sont toujours en état d'auto-échauffement et

souvent pyrophoriques. En plus des dangers d'incendie que présentent ces matériaux, il en est d'autres qui sont dus au sodium métallique utilisé dans la transformation du combustible, ainsi qu'au magnésium et au zinc fondus employés comme solvants dans diverses opérations. On élimine virtuellement ces risques en effectuant toutes les opérations dans une atmosphère inerte d'argon très pur. Le gaz de la cellule contient plus de 95% d'argon et normalement moins de 50 parties par million de vapeur d'eau et d'oxygène, et moins de 5% d'azote. L'appareil de purification du gaz de la cellule peut réduire l'oxygène et la vapeur d'eau dans l'atmosphère de la cellule à moins de 10 parties par million. Sans doute l'atmosphère de gaz inerte empêche-telle que les matériaux ne s'enflamment au cours du traitement, mais il faut prendre des précautions spéciales au moment des transferts dans la cellule ou hors de la cellule pour empêcher que cette atmosphère ne soit contaminée par de l'air et éviter que des incendies ne se produisent lorsque les matériaux pyrophoriques sont transférés dans les zones occupées par le personnel.

La sûreté nucléaire — prévention de la formation accidentelle d'un état critique — est l'un des problèmes les plus sérieux que posent les procédés aqueux, mais ne présente pas de réelles difficultés dans l'installation considérée. Les règles de sécurité de l'installation permettent de traiter en une seule fois 32 kg de combustible enrichi à 50% et de placer dans la cellule une quantité de combustible 16 fois plus grande. Cette quantité est supérieure à la charge totale d'un réacteur et est égale à environ sept fois le stock normal d'une usine. Le refroidissement de tous les fours se fait par rayonnement et non par circulation d'eau; par conséquent, non seulement on n'emploie pas de solutions aqueuses dans les opérations de traitement et de transformation, mais il n'y a aucune raison pour que de l'eau soit présente dans la cellule. De même, tous les fluides hydrauliques sont exclus, de sorte que la cellule ne contient aucun fluide modérateur. Des installations sont prévues pour le stockage en cours de traitement entre deux opérations ou en attente des résultats d'analyses.

A/280 США

Вопросы безопасности на установке по переработке облученного топлива реактора EBR-II

М. Левенсон *et al.*

Установка по переработке облученного топлива реактора-размножителя EBR-II будет использоваться для изучения пирометаллургических способов переработки облученного топлива и методов дистанционного управления этим процессом. Переработка облученного топлива, как и производство делящегося материала, связана с вопросами безопасности. Существующие затруднения увеличиваются во много раз, когда дело касается топлива с высоким уровнем выгорания, поступившим из реактора с высокой удельной мощностью, после охлаждения всего лишь в течение 15 дней. Компактные безынерционные системы, создание которых стало возможно благодаря развитию пирометаллургии, облегчают решение некоторых из этих задач. Основные вопросы, связанные с безопасностью такой системы, как установка по переработке топлива реактора EBR-II, могут быть разбиты на четыре группы: вопросы, связанные с защитой и непрерывным контролем полей излучения с целью предохранения рабочего персонала; вопросы, связанные с контролем радиоактивности и заражения; вопросы защиты от пожара и вопросы ядерной безопасности.

Защита установки спроектирована таким образом, что мощность дозы излучения на всех рабочих участках завода составляет меньше 0,1 *мр/ч*. Уровень излучения в камере установки будет равен 1 0000 000 *р/ч*. После того, как постройка камеры была закончена, была проведена радиография всей ее поверхности при помощи источника γ-излучения в виде Na^{24} и сурьмяно-бериллиевого источника нейтронов с целью проверки правильности сделанного расчета. На всех участках установки непрерывно измеряются фон γ-излучения и радиоактивность атмосферы. Воздух, выходящий из помещений, занятых установкой, находится под непрерывным дозиметрическим контролем для определения как суммарной мощности полей излучения, так и мощностей излучения отдельных видов излучения, активности J и Na^{24}. Газообразные отходы процесса переработки непрерывно контролируются для определения уровней излучения криптона и ксенона. Все эти данные автоматически записываются. При отклонении режимов работы от нормальных подается сигнал тревоги.

Продукты деления, поступающие на установку, разделяются на три потока. Первый поток содержит криптон, ксенон и тритий и выбрасывается в атмосферу. Применяемый дозиметрический контроль помогает установить, что нигде на установке или вне ее загрязнение воздуха не превышает установленных законом норм. Второй поток включает наибольшую часть продуктов распада благородных металлов. Эти продукты намеренно оставляются в топливе в виде примесей в сплавах и возвращаются в реактор. Третий поток (или, вернее, несколько твердых тел), содержащий иод, редкоземельные элементы и щелочноземельные металлы, герметизируются для распада и окончательного удаления (в твердой форме). Единственными жидкими отходами, содержащими продукты

распада, являются отходы, получаемые из аналитических лабораторий.

Металлические виды топлива, облученные до 20 000 *Мвт · сутки/т*, становятся самонагревающимися и часто самовоспламеняющимися. Кроме опасности пожара от таких веществ, существует также опасность пожара при употреблении металлического натрия в производстве тепловыделяющих элементов, а также растворителя жидкого магния и цинка, используемых на разных стадиях процесса. Все эти опасности фактически полностью устраняются тем, что вышеуказанные процессы проводятся целиком в инертной атмосфере аргона высокой степени чистоты. Газ в рабочей камере также состоит на 95% из аргона и, как правило, содержит меньше 0,005% водяного пара или кислорода и меньше 5% азота. Система очистки газа камеры способна понизить содержание кислорода и водяного пара в камере до уровня меньше 0,001%.

Хотя инертный газ и предотвращает возможность пожара во время производственного процесса, он требует соблюдения специальных мер предосторожности во время перевода обрабатываемого материала в камеру и из камеры, так как существует опасность загрязнения воздуха, а также возможность пожара после того, как самовоспламеняющийся материал переводится на другой участок, где работает персонал.

A/280 Estados Unidos de América

Aspectos relacionados con la seguridad de la instalación de tratamiento de combustibles irradiados del EBR-II

por M. Levenson *et al.*

La EBR-II Fuel Cycle Facility se utilizará para estudiar los métodos pirometalúrgicos de tratamiento del combustible irradiado y los métodos para la refabricación a distancia del combustible. En el tratamiento de combustible irradiado y en la fabricación de elementos con material fisible se plantean siempre importantes problemas de seguridad. Muchos de estos problemas adquieren aun mayor importancia cuando el material es combustible que ha experimentado un alto grado de quemado, procedente de un reactor de gran potencia específica y que se ha dejado enfriar únicamente durante 15 días. Por otra parte, los sistemas compactos no moderados, que hace posibles la pirometalurgia hacen algunos problemas menos difíciles. Los problemas básicos de seguridad de la instalación EBR-II Fuel Cycle Facility pueden dividirse en cuatro grupos: blindaje y vigilancia radiológica para la protección del personal, control de la radiactividad y contaminación, medidas contra incendios y seguridad nuclear.

El blindaje de la EBR-II Fuel Cycle Facility se proyectó para reducir la radiación a menos de 0,1 milliroentgen/hora en todas las zonas de trabajo de la instalación. La radiación en el interior de la celda será de 1 000 000 roentgens/hora. Después de terminada su construcción se hicieron radiografías de toda la estructura de la celda utilizando una fuente de radiación gamma de sodio-24 y una fuente de neutrones de antimonio/berilio para comprobar la idoneidad del proyecto. Todas las zonas de trabajo de la instalación se vigilan constantemente con registros de la actividad gamma de fondo y la atmosférica. Se registra de manera continua la actividad total y las actividades correspondientes a partículas, a sodio-24 y a yodo de todo el aire evacuado de la instalación. El gas expulsado de la zona de tratamiento se examina constantemente respecto a las actividades de kriptón y xenón. Todos los datos de actividades radiológicas recogidos se registran automáticamente y, en caso de situación anormal, entran en acción sistemas de alarma.

Los productos de fisión que entran en la instalación se dividen en tres grupos. El primero, que contiene kriptón, xenón y tritio, se evacúa a la atmósfera en forma controlada para asegurarse de que en ningún punto, dentro o fuera de la instalación, se sobrepasan los límites máximos permitidos. El segundo, que contiene la mayoría de los productos de fisión correspondientes a los metales nobles, se mantiene intencionadamente en el combustible (como componentes de aleación) y se devuelven con éste al reactor. El tercero (en realidad varios productos sólidos producidos directamente) que contiene yodo, tierras raras y metales alcalinotérreos, se almacena para que vayan desintegrándose y finalmente proceder a su eliminación (como sólidos). Los únicos residuos acuosos líquidos que en funcionamiento normal contienen productos de fisión, son los procedentes de los laboratorios de análisis.

Los combustibles metálicos irradiados hasta 20 000 MWd/t están sometidos en todos los casos a un proceso de autocalentamiento y son, en muchos casos, pirofóricos. Además del riesgo de incendio que presentan tales materiales, existen también los riesgos de incendio debidos al sodio metálico, utilizado en la fabricación de combustible, y del magnesio fundido y disolventes del cinc, que se utilizan en diversas operaciones del proceso. Estos riesgos se eliminan prácticamente realizando todas las operaciones en una atmósfera inerte de argón muy puro. El gas de las celdas contiene más de 95% de argón y normalmente menos de 50 ppm de vapor de agua y oxígeno y menos de 5% de nitrógeno. El sistema de purificación de dicho gas puede reducir la concentración de oxígeno y vapor de agua en la atmósfera de la celda a menos de 10 ppm. Aunque la atmósfera inerte impide la inflamación de los materiales durante el proceso, se necesitan precauciones especiales en las operaciones de

entrada y salida de los mismos en la celda con objeto de evitar la contaminación con aire y la producción de fuegos después de haber trasladado los materiales pirofóricos a sectores ocupados.

La seguridad nuclear — prevención de criticidad inadvertida — es uno de los problemas más importantes en los procesos acuosos, pero en esta instalación es únicamente un problema nominal. Las normas de seguridad de esta instalación permiten tratar por operación 32 kg de combustible con 50% de enriquecimiento; 16 veces esa cantidad, lo que representa más de la carga total del reactor y aproximadamente unas siete veces el inventario normal de la instalación, puede encontrarse simultáneamente en la celda. Todos los hornos están proyectados para enfriamiento por radiación y no para refrigeración por agua, de este modo no sólo no se utilizan soluciones acuosas durante el tratamiento o fabricación, sino que no se utiliza agua en la celda para ninguna operación. Igualmente se excluyen todos los fluidos hidráulicos de modo que no hay líquidos moderadores en la celda. Dispositivos muy flexibles permiten el almacenamiento entre diversas fases del tratamiento o mientras se esperan resultados analíticos.

Safety and siting considerations in the design of the NFS power fuel reprocessing plant

By T. C. Runion and W. A. Rodger *

Nuclear Fuel Services, Inc., a subsidiary of W. R. Grace & Company, is building, in western New York State, the first privately owned plant for the chemical processing of spent reactor fuel from the nuclear power industry. Because it is the first such privately owned plant a number of new safety and siting problems had to be considered. Some of the more interesting are discussed here. A complete safety analysis and plant description is available elsewhere [1].

The NFS plant is a multi-purpose plant capable of processing any type of nuclear fuel element from which the fuel proper can be reduced to a nitric acid solution. No fuel will be processed before it has been cooled for 150 days. The base line process is a chop-leach head-end with Purex solvent extraction for processing 1 000 kg/day of low enriched UO_2 in stainless steel or zirconium alloy tubes. In addition, there will be equipment for the dissolution of zirconium-uranium alloy fuels in a HNO_3-HF mixture, and nitric acid dissolution of uranium and aluminium-uranium or molybdenum-uranium alloy fuels.

PLANT SITE

Under existing US law it is necessary to carry out the storage or burial of radioactive waste upon land which is owned by either the federal or a state government. Since it does not appear as yet to be economical to store the wastes from a chemical reprocessing plant at other than its own site, this law requires, in effect that the plant be located on government owned land.

The early development of the nuclear power industry in the US has been largely in the high-fuel-cost northeast with a second area on the west coast following along a little behind. To best service the early load at lowest shipping cost the choice of a northeast location seemed indicated.

After considering several locations, a site in the Western New York Service Center was chosen. The Center, consisting of 3 331 acres in Cattaraugus County, New York, had been set aside by the State for the burial of low-level radioactive wastes. It is State owned and under the jurisdication of the New York State Atomic and Space Authority.

The specific site requirements for chemical processing are, of course, much more stringent than those for low-level burial. It was necessary, therefore, to extend and intensify the site study which had already been carried out by New York State in order to determine the suitability of this particular site for chemical processing.

Surrounding population

The site is located in an area of low population density (Table 1). Only 25 per cent of the area within a 25-mile radius exceeds a population density of 60 per square mile. The nearest village, Springville, 4½ miles to the north, has a population of 3 852. There is no city or village within 25 miles with a population greater than 10 000.

Table 1. Total population within successive 5-mile radii of the Western New York nuclear service center

Radius, miles	Total population, persons	Population density, persons/square mile
5	5 765[a]	73
5	2 685[b]	36
10	15 644	49.6
15	39 297	55.5
20	62 432	49.5
25	170 573	87

[a] With Springville. [b] Without Springville.

Meteorology

The climatology of this site is characterized by considerable rainfall and humidity and rather wide temperature extremes. The area has a mean annual temperature of 45 °F, an annual rainfall of 40 inches, and an annual snowfall of 80 to 100 inches. The prevailing wind directions are from the northwest in the winter

* Nuclear Fuel Services, Inc., Washington, DC.

and from the southwest in the summer. The area around the site is seldom subject to persistent stagnant high pressure areas and poor diffusion conditions. The assumptions used for all of the diffusion calculations for the initial hazard analysis are shown in Table 2 along with the wind roses predicted from surrounding stations and those which have been subsequently measured at the site.

Table 2. Probable wind roses and diffusion parameters for the Western New York nuclear service center [a, b]

	Wind roses (per cent of time)			
	Summer		Winter	
	Predicted	Measured	Predicted	Measured
S . . .	30	22.6	15	19.3
SW . .	25	13.1	15	24.0
W . . .	8	8.8	8	14.4
NW . .	15	19.8	35	23.3
N . . .	4	9.3	6	9.1
NE. . .	3	4.7	3	2.0
E . . .	5	4.8	3	2.1
SE . . .	10	16.9	15	5.8

	Diffusion parameters for use when conditions being studied are expected to persist for:				
	100 days	3 hours to 1 day		Instantaneous	
		Inversion	Lapse	Inversion	Lapse
n. . .	0.25	0.5	0.25	0.5	0.25
C_y	0.4	0.4	0.5	0.2	0.25
C_z	0.4	0.07	0.2	0.07	0.25
C_x	—	—	—	0.2	0.25
u, m/s	4	1	4	1	4

[a] Memo, M. E. Smith to W. A. Rodger, 23 May 1962, estimated from general meteorological conditions as modified by the local terrain.

[b] Wind rose data taken during 1963 on-site are compared with the original predictions.

Geology

The geology of the site has been investigated by a series of drillings and a seismic survey. Bedrock, which has a profile steeper than the surface, is overlain by glacial deposits. The surface layer on part of the site is relatively permeable coarse granular deposits, the thickness of which varies from 0 to 25 feet. This is underlain by a nearly completely impermeable silty till varying in thickness from 25 to 35 feet. On part of the site, this silty till is at the surface. Under that there may be irregular sandy tills and various shales. All of these layers have been shown to have good ion exchange capacities.

Ground water

Usable quantities of ground water occur in three aquifers on the site. The term "usable" may here be construed as a yield of 1 to 10 gallons per minute from a well or spring. The coarse granular deposits at the surface constitute the uppermost aquifer in the construction area. This aquifer is referred to as the nonartesian aquifer.

In much of the site the stream valleys cut completely through this aquifer. An important result of this segmentation is the relatively short distance of travel by the ground water from points of recharge to points of discharge into surface streams. The rate of groundwater movement is determined by the hydraulic gradient and the permeability and porosity of the aquifer. Under natural conditions, the velocity of ground water in the coarse granular deposits is in the range of 1 to 2 feet per day.

The silty clay till in the construction area is not an aquifer. It is, however, largely water-saturated. The movement of ground water through the till is very slow and almost entirely capillary in nature. Samples were submitted to the Hydrologic Laboratory of the US Geological Survey for determination of porosity, permeability and grain-size analysis. For samples at 3 to 5 feet, 13 to 14.8 feet and 45 to 47 feet the permeabilities were found to be 0.002, 0.002, and 0.001 g/d ft^2 in the vertical direction and 0.005, 0.004, and 0.004 g/d ft^2 in the horizontal direction, respectively. With these permeabilities it would take water approximately 40 000 years to work its way through the 700 feet from the plant and waste tank area to the nearest ravine.

The thin sand unit beneath the uppermost till sheet constitutes a second aquifer in the valley area. This aquifer is referred to as the shallow artesian aquifer. Certain of the data suggest that this sand unit may be completely enclosed by the till sheets and that the contained water may be controlled by capillary phenomena.

The shale bedrock comprises the third aquifer in the area. This aquifer is referred to as the bedrock artesian aquifer. The rate and direction of movements of water in this aquifer are essentially unknown because of the few wells available for observation.

The particular area chosen for the plant facilities is one in which the artesian aquifer is missing. This area is a "peninsula" bordered on the east by Erdman Brook and on the west by Quarry Creek. It can be definitely stated that any activity getting into the ground water in the plant site area will show up eventually in one of these two streams and nowhere else except, of course, for that which is sorbed upon the soils and held therein. This situation allows the establishment of an excellent monitoring system with a minimum of expense and a maximum of reliability.

Surface water

All of the surface drainage from the entire site, as well as the drainage from the upper layers of till, is collected in Buttermilk Creek through a series of small tributaries which include Erdman Brook and Quarry

Creek. At the northwest corner of the site, Buttermilk Creek joins a larger stream, Cattaraugus Creek. The confluence of these two streams and approximately five miles of Buttermilk Creek are included within the site boundaries. Cattaraugus Creek empties into Lake Erie 27 miles west of Buffalo. The average daily stream flows over a 15-year period have been measured or calculated as follows: Cattaraugus Creek at Gowanda (19 stream miles west of site), 704 ft^3/s; Cattaraugus Creek at Buttermilk Creek, 358 ft^3/s, and Buttermilk Creek, 41 ft^3/s.

There are no villages or cities downstream from the site which rely on Cattaraugus Creek for their water supply and there are no potable water sources or large water supplies used in the immediate area. Lake Erie is used as a source of public water, but there is no such use closer than 7 miles from the mouth of the Cattaraugus. There is no commercial or industrial use of water on or within 4 miles of the site. All existing water supplies are obtained from shallow, large-diameter dug wells or springs.

Seismic data

Western New York is an area of low seismic activity. According to the records of the US Coast and Geodetic Survey, in more than 100 years there has been only one earthquake classified as "destructive or near destructive" within a radius of 200 miles of the site. Detailed records since 1928 list only six earthquakes with epicenters within 75 miles of the site. Of these only the Attica earthquake of 12 August 1929 exceeded Intensity V. The nearest fault is at a distance of 35 to 40 miles from the site and even this structure cannot be completely proved as a fault.

SAFETY ANALYSIS OF PLANT DISCHARGES

An analysis of the operation of this plant, both that to be expected normally and certain unlikely abnormal events, has been made to show that in all cases it may be operated within this particular environment in such a manner as to meet all the legal requirements for a radiochemical plant.

Radioactivity can be lost from the process complex at the following points: (*a*) stack, (*b*) waste storage tanks, (*c*) liquid discharge, (*d*) burial ground. Each of these points will be discussed to show that under the normal operation of this plant there will be no releases of radioactivity to the environment in excess of the legal limits.

In addition a number of unlikely abnormal events have been hypothesized. The rationale for the selection of these events has been to select for the plant, the stack, and the tank farm, events which represent the upper limit of catastrophe which could occur in each of these areas, even though we believe that the likelihood of occurrence is very small.

Normal operation

During normal operation the only discharge of activity to the environment would be via ventilation air from the stack, low-level liquid waste discharge, and solids burial. Calculations on these discharges are based upon an assumed average activity content from a one ton/day of fuel specified by the following parameters: 20 000 MWd/ton burn-up, 27.5 MW/ton specific power, 2-year irradiation at 85 per cent load factor and 150 days cooling before processing.

Stack discharge

The ventilation systems are designed to assure that, under normal operation conditions, flow of air is always from areas of least contamination into those of higher contamination. There are separate systems for vessels, dissolvers, and the cells themselves. These join together and are filtered before discharge through a 65-meter stack. Iodine removal facilities are designed to collect 99.5% of the incident iodine. It is assumed that all of the noble gases in a daily charge escape during the course of the day. Under normal operating conditions, the amount of solid fission products taken into the gas stream is assumed to be low enough that the filtering of this stream will reduce them to the point where they are negligible in comparison to the gaseous activity. Under these conditions, the total daily discharge from the stack will be: ^{85}Kr, 8.25×10^3 curies; ^{129}I, 8.5×10^{-5} curie; ^{131}I, 8.5×10^{-3} curie; $^{131}Xe^m$, 1.1 curies; ^{133}Xe, 5.0×10^{-3} curie; tritium, 3.1×10^2 curies.

The concentration of each of these isotopes has been calculated for both inversion and average conditions over the range 1 500 to 51 000 meters. The results of these calculations are presented in Table 3. The

Table 3. Calculated maximum concentration of gaseous isotopes under inversion[a] and average[b] meteorological conditions

Isotopes[d]	Curies/second	Inversion	Average	MPC[c] ($\mu c/cm^3$)
Kr . . .	9.5×10^{-2}	9.5×10^{-7}	2.1×10^{-8}	3×10^{-7}
^{129}I . . .	1×10^{-9}	1×10^{-14}	2.2×10^{-16}	6×10^{-11}
^{131}I . . .	1×10^{-7}	1×10^{-12}	2.2×10^{-14}	1×10^{-10}
$^{131}Xe^m$.	1.3×10^{-5}	1.3×10^{-10}	2.9×10^{-12}	4×10^{-7}
^{133}Xe . .	5.8×10^{-8}	5.8×10^{-13}	1.3×10^{-14}	3×10^{-7}
Tritium[e] .	3.6×10^{-3}	3.6×10^{-8}	8×10^{-10}	2×10^{-7}

[a] Maximum concentration occurs at about 6 000 meters from the stack; concentration within about 10% of the maximum occurs from about 4 000 to 10 000 meters from the stack.

[b] Maximum concentration occurs at the site boundary (1 500 meters).

[c] Table 2, Appendix B, 10 CFR Part 20.

[d] At 150 days' cooling these are the only significant gaseous isotopes.

[e] All 310 curies up the stack—only about 75 curies are expected to escape by this route.

maximum concentrations are given for both the average and inversion conditions. For average conditions the maximum concentration occurs at the site boundary; under inversion conditions the maximum concentration occurs over the range of about 4 000 to 10 000 meters downwind from the stack. It can be seen that all of the concentrations are well wilhin the MPC values with the exception of the ^{85}Kr concentration under inversion conditions. The inversion concentrations given are centerline concentrations and include no wind diversity factor; they are not expected to persist for more than a few hours at a time. Thus, the yearly average concentration will not be significantly increased by these occurrences.

The plant is designed to release iodine at a concentration low enough to protect those areas surrounding the plant site which are used for dairying. Using the long-period average concentration and a deposition velocity of 0.01 meter per second, the deposition rate has been calculated. Since yearly average concentrations are used, it is reasonable to assume that the equilibrium conditions are reached; i.e., the rate of deposition equals the rate of decay. A wind frequency of 25 per cent per octant has been used in these calculations. It was found after the Windscale incident that a grazing area contamination level of 1 μc per square meter resulted in about 0.1 μc/liter of milk [2]. Using this relationship the resultant activity levels in milk have been calculated and are shown in Table 4.

Table 4. Calculated iodine deposition and milk concentration

Distance (m)	Ground concentration (pc/m²)	Milk concentration (pc/ℓ)
1 500	200	22
2 000	150	15
5 000	31	3.1
10 000	8.9	0.89
20 000	2.6	0.26

The Federal Radiation Council has established a *Radioactivity Intake Guide for Iodine-131* of 100 pc per day, based on the uptake by children as the most sensitive segment of the population. As can be seen from Table 4 the consumption of about five liters of milk per day from dairy cattle grazing immediately adjacent to the site boundary would be required to equal the level of intake as established by this guide. This rate of consumption is higher than any that can be expected, probably by a factor of at least four. In addition, no credit is taken for dilution (during processing) by milk containing lesser (or no) amounts of radioiodine nor for the fact that open pasturage is used only about 7 months out of the year in this area.

Liquid waste

High-level liquid waste is stored on site in waste tanks. There would be no discharge from these in normal operation. The very-low-level wastes from the process—overheads from acid fractionation, solvent wastes, and miscellaneous wastes—can be put through a general purpose evaporator and the overheads from this can be put through ion exchange columns if necessary. It is expected that with the exception of tritium the normal activity content of the overheads will contain about 10^{-6} μc/cm³ of activity. This can be further reduced a factor of 30 by the use of simple, non-regenerated cationic ion exchange, resulting in a concentration of 3×10^{-8} μc/cm³. The expected volume of these wastes is 10 000 g/d. The average available flow in Buttermilk Creek is 41 ft³/s which is equal to 2.7×10^{7} g/d. Thus, the available on-site dilution factor is 2.7×10^{3}. In Cattaraugus Creek an additional dilution factor of 8.5 is available. The concentration in Cattaraugus Creek would then be expected to be about 10^{-11} μc/cm³. Furthermore, the residual activity in this stream will be largely ^{106}Ru and ^{131}I with some ^{95}Zr-^{95}Nb. The MPCs for these isotopes are 1×10^{-5}, 2×10^{-6}, 6×10^{-5}, and 1×10^{-4} μc/cm³. Therefore, the available factor of safety is about 10^{4} without any analyses of the effluent and about 10^{5} if we choose to carry out specific fission-product analyses on this effluent stream.

Tritium presents a problem since there is no way to remove it from the aqueous waste streams once it is present therein as tritiated water. We expect that about 75% of the tritium in the fuel elements will find its way into the storage lagoons and be discharged into the creek systems. A maximum concentration of tritium in Cattaraugus Creek of about 2.5×10^{-4} μc/cm³ is expected compared to an MPC for this isotope of 3×10^{-3} μc/cm³. It is clear that the analytical techniques used in determining our effluent concentrations will have to discriminate between tritium and other fission-product activities.

Thus, this low-level stream can be discharged directly to Buttermilk Creek and the level of activity at the site boundary will remain well within the MPC levels of 10 CFR Part 20. In addition, there will be a lagoon system available for use as an emergency hold-up area. Its use will permit time for the decay of shorter-lived isotopes and will allow the absorption on the soil of some of the longer-lived isotopes.

Solid waste

Two types of solid waste will be buried on the site. One is low-level solid trash both from the operation of the plant and from other sites. In this material the radioactivity is very low, estimated to be about 0.01 curie/cubic foot and defined as less than 1 curie/

cubic foot, but is assumed to be in a mobile condition. The second, made up of hulls from the dissolver, contains very considerable amounts of radioactivity in the form of induced activity in the stainless or zirconium hulls, estimated to be about 100 000 curies/cubic foot, but it is considered immobile. Both will be buried directly in the silty till. Because of the very low permeability of this till any activity which was leached would not be expected to move significantly from the point of burial. Finally the silty till has been shown to have ion exchange capacity and any leached activity could be expected to be absorbed upon the surrounding soil before it could appear at a surface.

Abnormal operations

Four abnormal events ranging from the unlikely to the incredible have been hypothesized to delineate, we believe, the upper limit effects of any catastrophe which could occur in this plant and its related facilities. These are: (*a*) the complete rupture of a waste tank releasing 600 000 gallons of high-level waste, (*b*) a criticality incident anywhere in the plant involving a total of 10^{19} fissions in a single burst or a multiple continuing event totalling 10^{20} fissions, (*c*) a chemical explosion in the plant which is assumed to rupture a vessel containing a full day's charge of the maximum fission-product content possible, and (*d*) the complete failure of the iodine removal equipment so that for a period of up to one day the complete charge of iodine is lost to the atmosphere. Each of these is analyzed to show that the plant environment would be capable of accepting these unlikely events.

The analysis shows that none of these accidents would result in levels of exposure to the general public exceeding the guide limits for gaseous emission suggested in Section 100.00 of 10 CFR Part 100; and further there is reasonable assurance that liquid discharges at the site boundary would be kept within the concentrations for drinking water purposes specified in 10 CFR Part 20.

Waste tank rupture

Careful measures have been taken to ensure the reliability of the high-level waste tanks, to provide multiple means of detecting any leakage in the unlikely event that any defects should develop and to minimize the effects on the environment of such leakage.

The waste storage tanks are built in a cup-and-saucer design. Operating procedures call for monitoring of the annular space between the tank and its saucer. Both the tank and its saucer are contained inside a concrete vault which affords additional protection. The entire tank-vault complex is located in the highly impermeable silty till already described. In order to maintain the bearing qualities of the silty till it is necessary to maintain its natural moisture content notwithstanding the heat loss from the tank. Because of the impermeability of the silty till, moisture must be deliberately introduced under the tank. This is done by constructing the tank-vault complex on a four-foot layer of graded gravel and introducing water into this area. Standpipes built down to this area afford another monitoring point. The area also provides a secondary saucer from which escaped radioactivity could be recovered. Thus, there are five separate barriers between the waste and the environment and under normal operation *no* escape of radioactivity is expected.

In the unlikely event a tank saucer and vault should rupture, the first step would be to pump out the remaining contents of the tank to a spare tank which is always available. If the normal piping system has been made inoperative by the same event which resulted in rupture of the waste tank, a temporary pump and a temporary line could be established to accomplish the transfer of the tank contents. This system could also be used, by installing a temporary line in a standpipe, to pump the solution in the sub-tank gravel into the spare tank. Allowing three days to set up the temporary piping system and seven days for pumping would mean that approximately ten days maximum would have passed before the tank contents were recovered and transferred.

The silty till has been shown to be so impermeable that it could be expected to act essentially as a tank itself during the recovery period. Thus, it is expected that even this most drastic accident could be coped with in a manner so as to protect the public completely.

In any event there still remains the protection of the specific site location. Studies have shown that if any activity does escape through or over the surface of this land, it must appear at the confluence of Erdman Brook and Quarry Creek. A monitor station will be maintained at this point. Any loss would be detected here. The flow of water at this point, an average of 2 ft^3/s is not so great but that steps could be taken to process it if this became absolutely necessary.

Criticality accident

The assumption is made here that a criticality incident producing 10^{19} fissions in a single burst or 10^{20} fissions in a repeating incident is experienced and that the entire production of gaseous fission products (from 10^{20} fissions) is lost.

It is well established both from theoretical calculations [3] and experience [4] that the limiting off-site condition in an occurrence of this type is the thyroid dose from the iodine isotopes released. Table 5 lists the peak activity of each of the iodine isotopes 131 through 135 and the time after the accident when the peak occurs [5]. This event is analyzed for average and inversion conditions on the basis of thyroid dose to a

Table 5. Quantities of iodine isotopes formed from 10^{20} fissions

Isotope	Time of peak activity	Peak activity, curies
^{131}I	5.2 hours	73
^{132}I	7.6 hours	240
^{133}I	3.5 hours	1 250
^{134}I	46.0 minutes	17 300
^{135}I	2.2 minutes	4 400
	Total	23 300

person on the periphery of the site (1 500 meters), at Springville (7 200 meters), and at Buffalo (51 000 meters). With one exception, the peak activities have been assumed in calculating the population dose. The one exception is the activity of iodine-134 at the time it reaches Buffalo under inversion conditions. The transit time in this case is so large in relation to the half-life of iodine-134 and its precursors that its activity level was found to be negligible compared to the remaining iodine isotopes.

Using the approximations suggested in 10 CFR Part 100 for the thyroid dose from each of these isotopes, the total rem per person and the fraction of the 300 rem reference value are calculated for the three locations and for both types of meteorology. Total man-rem values have also been calculated. These data are presented in Table 6. It can be seen that in no case is the reference value used for evaluation of reactor sites exceeded or even closely approached. The highest value indicated, a 5.85-rem/person dose in Springville under inversion conditions, is not expected to be encountered since it is the opinion of meteorologists that an inversion aimed at Springville would be caught and held in the Buttermilk-Cattaraugus Valley systems.

Table 6. Individual and population doses at several points in event of a criticality incident

	Individual dose, rem/person	Fraction of 300 rem/person dose	Exposed population dose, rem total
Site periphery			
Inversion[a]	0.28	9.3×10^{-4}	—
Average[a]	2.5	8.3×10^{-3}	—
Springville			
Inversion[a]	5.85[b]	1.9×10^{-2}[b]	73[b]
Average[a]	0.19	6.3×10^{-4}	2.4
Buffalo			
Inversion[a]	1[b]	3.3×10^{-3}[b]	1750[b]
Average[a]	8.4×10^{-3}	2.8×10^{-5}	15

[a] See Table 2 for conditions assumed.

[b] It is not expected that, under inversion conditions, any activity will reach Springville or Buffalo; but rather will be trapped in Cattaraugus Valley.

Chemical explosion in cell

The assumption is next made that a vessel containing one full day's charge of fuel in solution suffers an explosion which ruptures the vessel distributing the contents throughout a cell and putting some fraction of the contained solution into the ventilating system. An analysis of a completely analogous situation has been made by ORNL [6]. The assumptions are made therein that aerosols penetrating high-efficiency filters will contain 0.15 mg/m^3 of air with the same concentration as the original dispersed solution and that the MPC for fixed fission products is 6.6×10^{-9} μc/cm^3. The ventilating air passing through the filters of this plant amounts to 32 000 ft^3/min or 900 m^3/min. This would result in a release of 2.5 g of solution assuming that in twenty minutes the ventilating system will have picked up nearly all of the gross activity that it is going to. The maximum activity to be expected in the plant is about 0.45 curie per gram for a total discharge of 1.1 curies. The highest value (at a distance of about 5 000 meters) for ground concentration in this case is 3×10^{-9} μc/cm^3. This is less than the MPC for mixed fission products assumed and it would appear possible to accept this particularly untoward accident.

Failure of iodine removal equipment

Finally it is assumed that the silver reactors and other iodine removal equipment all fail and that this is not discovered for a period of one day. It is extremely unlikely that the failure could remain undetected for 24 hours since the stack monitor would detect the iodine increase at once. If the entire charge of iodine-131, 1.7 curies, were to be lost during a day under worst conditions (at a distance of about 5 000 meters) the concentration of iodine-131 would be 2×10^{-10} μc/cm^3. This is twice the new MPC for continuous exposure off-site.

CONCLUSIONS

From the foregoing discussion it may be concluded that the limiting aspects in the operation of this plant in relation to its particular environment are as follows:

Normal operation		
Stack discharge	^{85}Kr	Breathing and immersion
	^{131}I	Deposition on pasturage and uptake in milk
Liquid discharge	Tritium	MPC for unrestricted use
Abnormal operation		
Criticality	Iodines	Thyroid dose

It has been shown that the environment of this plant is capable of accepting the amount of waste products

predicted for it without exceeding established safety limits. The long-lived limiting isotopes listed above (^{85}Kr and tritium) cannot be reduced by any reasonable decay period. These, then, may be expected to eventually dictate the upper limits of processing rate at which this plant, or for that matter any chemical reprocessing plant, may operate.

Iodine-131, on the other hand, is readily reduced by storage prior to processing. Fuel cannot be shipped into the plant until it has been cooled for 90 days. For most shipments a minimum of 30 days will be required and often a good deal more. To assure that iodine deposition and uptake into milk will be adequately low, it has been decided to provide at least 150 days cooling prior to any processing.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/281 Etats-Unis d'Amérique

Considérations de sûreté et de choix du site dans l'étude d'une usine de traitement du combustible irradié

par T. C. Runion et W. A. Rodger

Lors de l'étude de la première usine privée de traitement du combustible irradié que construit actuellement la société Nuclear Fuel Services Inc., à West Valley (Cattaraugus County, New York), il a fallu tenir compte de plusieurs exigences nouvelles en matière de sûreté et de choix du site. Cette usine est la première de ce genre qui soit soumise à la procédure d'autorisation de la Commission de l'énergie atomique des Etats-Unis, et, pour cette raison, sa création a posé un certain nombre de problèmes. Conformément à la législation actuelle des Etats-Unis, cette usine devait être construite sur un terrain appartenant soit à un Etat, soit au Gouvernement fédéral. Il était souhaitable, pour des raisons économiques, que l'usine fût située dans la partie nord-est des Etats-Unis, près du centre géographique des réacteurs qui donneraient à traiter les premières charges de combustible irradié. L'usine se trouverait ainsi dans une région relativement humide et il fallait envisager avec soin les problèmes que posent les précipitations, les nappes aquifères et l'utilisation des eaux.

L'Etat de New York avait choisi un emplacement aux fins d'enfouissement des déchets nucléaires de faible activité. Une étude approfondie des caractéristiques géographiques, météorologiques, géologiques, hydrologiques et séismologiques de cet emplacement ont permis de conclure qu'il convenait également à l'exploitation d'une usine de traitement du combustible irradié.

Les auteurs examinent en détail les caractéristiques pertinentes de cet emplacement ainsi que les rapports entre l'usine et son environnement, tant dans les conditions normales d'exploitation que dans les cas d'anomalies possibles du fonctionnement. Ils indiquent les limites que l'environnement impose aux caractéristiques de l'usine.

Lorsque l'on a étudié l'usine en fonction de son emplacement, on a fait des recherches sur des problèmes de sûreté particuliers au nombre desquels figurent: le temps de désactivation pour la décroissance des radioisotopes à courte période, la conception de réservoirs de stockage compatibles avec les lieux d'implantation, les précautions à prendre pour prévenir un état critique, etc.

A/281 США

Обсуждение вопросов безопасности и размещения при проектировании завода фирмы „Ньюклеар фьюэл сервисиз“ по переработке облученного топлива энергетических реакторов

Т. Ч. Руниoн, У. А. Роджер

При сооружении фирмой «Ньюклеар фьюэл сервисиз» первой частной установки по переработке облученного топлива в Уэст Вэлли (графство Каттараугус, шт. Нью-Йорк) необходимо было учитывать целый ряд новых правил и постановлений, относящихся к безопасности и выбору местоположения такого предприятия. Поскольку это был первый частный завод по переработке облученного топлива в стране и для его постройки требовалось разрешение Комиссии по атомной энергии США, перед фирмой встали вопросы, которые приходилось решать впервые.

Во-первых, по закону такая установка должна быть построена на участке земли, принадле-

жащем либо штату, либо федеральному правительству. По экономическим соображениям считалось желательным расположить завод в северо-восточной части страны, чтобы он находился вблизи географического центра установок, являющихся источником облученного топлива. Это заставило построить завод в районе сравнительно высокой влажности, где пришлось тщательно учитывать количество атмосферных осадков, уровни грунтовых вод и количество употребляемой воды.

Власти штата Нью-Йорка уже раньше выбрали место для захоронения малоактивных отходов. Интенсивное изучение географических, метеорологических, геологических, гидрологических и сейсмологических характеристик этого места привело к заключению, что оно также удобно для постройки завода по переработке облученного топлива.

Доклад содержит подробное обсуждение соответствующих характеристик данной местности. Описано влияние нового завода на окружающую среду как при нормальной его работе, так и в случае какой-либо аварии. Указаны факторы, ограничивающие параметры завода с точки зрения его влияния на окружающую среду.

Из специальных вопросов безопасности, которые приходилось принимать во внимание при проектировании настоящего завода, можно назвать следующие: время для распада изотопов, имеющих короткий период распада; задачи, связанные с механизацией начальных процессов; конструкция баков для хранения топлива, соответствующих местным условиям, и соображения критичности. Каждый из этих вопросов обсуждается в докладе.

A/281 Estados Unidos de América

Consideraciones sobre la seguridad y el emplazamiento en el proyecto de la instalación de tratamiento de combustibles irradiados de la NFS

por T. C. Runion y W. A. Rodger

Al establecerse la primera instalación de tratamiento de combustibles irradiados de propiedad privada (que está construyendo Nuclear Fuel Services, Inc., en West Valley, Cattaraugus County, Nueva York) era necesario considerar una serie de nuevos requisitos sobre seguridad y emplazamiento. Esta es la primera instalación de su clase sometida al procedimiento de autorización de la USAEC de modo que, por vez primera, hubo de centrar y someter a formalidades públicas un buen número de problemas. De acuerdo con las leyes vigentes en EE.UU. era necesario situar esta instalación en terrenos de propiedad del Estado o de propiedad federal. Por razones económicas era conveniente situar la instalación en la parte noreste de los EE.UU. próxima al centro geográfico de la zona que próximamente producirá, a causa de los reactores allí instalados, combustibles irradiados para tratar. Este hecho sitúa la instalación en una zona relativamente húmeda, donde los problemas de las lluvias, la capa freática y la utilización del agua tienen que ser considerados cuidadosamente.

El Estado de Nueva York había seleccionado un emplazamiento para realizar el enterramiento de residuos radiactivos de baja actividad. Un estudio detallado de las características geográficas, meteorológicas, geológicas, hidrológicas y sismológicas del emplazamiento llevó a la conclusión de que era también un lugar adecuado para el funcionamiento de una instalación de tratamiento de combustible irradiado.

En la memoria se presenta una discusión detallada de las características propias de este emplazamiento. Se discute también la relación de la instalación con los alrededores, tanto en condiciones de funcionamiento normal como en relación con posibles anormalidades de funcionamiento. Se señalan las características limitativas del proyecto y explotación de la instalación en relación con el medio ambiente.

Algunos de los problemas especiales de seguridad que se consideraron en el proyecto de la instalación en relación con su emplazamiento son: tiempo de enfriamiento para la desintegración de isótopos de vida corta, los problemas particulares asociados con los tratamientos mecánicos previos, el diseño de tanques de almacenamiento compatibles con la situación local y requisitos de criticidad. Se hace una discusión de cada uno de estos aspectos.

Guide for the hazards evaluation of a nuclear chemical installation

By J. Tadmor and H. Galron *

Experience acquired in operating nuclear installations shows that in spite of the potential hazards of the nuclear industry, its safety record ranks amongst the highest prevailing in the industrial field. This result has been achieved by the incomparable efforts made to recognize the hazards of nuclear installations, to avoid occurrence of failures and accidents and to minimize the possible effects of an accident, should it occur.

The hazards evaluation study made prior to the construction and operation of a nuclear plant, which has been adopted as a standard procedure, is an example of the particular efforts made in the nuclear safety field. The hazards evaluation study is compiled in the form of a report which includes full data on the enterprise itself, an evaluation of all the hazards possibilities, however slight, and the resulting effects on the installation and its surroundings, should an accident occur.

The hazards evaluation study and compilation of the report have a threefold objective:

(*a*) To enable the planners and operators of the plants to recognize the critical aspects of the installation from the safety standpoint;

(*b*) To facilitate evaluation by the Licensing Commission or inspection teams of potential dangers associated with operation of the plants, and the consequences of credible accidents;

(*c*) To provide basic information which will assist especially new technical employees in learning about the enterprise and the hazards involved.

The numerous reports published on the hazards evaluation of nuclear reactors as well as the guides for such studies (see, for example, Refs. [1, 2]) point out the general safety problems encountered in the operation of these plants. The position is not analogous as regards nuclear chemical plants or laboratories undertaking work with radioactive materials. Few hazards evaluation reports [3, 4] have been published for nuclear chemical installations and they cannot be taken as general guides as regards the topics to be studied and included in hazards evaluation reports for different installations, because of the variety of the safety problems which may be encountered.

* Israel Atomic Energy Commission.

However, the hazards involved in the operation of these installations may in some cases exceed those met with in the operation of nuclear reactors. Thus, a hazards evaluation study performed by Guthrie and Nichols [5] for a typical fuel fabrication and radioisotope processing plant reveals that the worst uncontained accident, involving a release of 350 g of ^{90}Sr, could lead to a potential economic loss of $\$5 \times 10^9$, while that involving a release of 2 kg of ^{238}Pu could lead to a potential economic loss of $\$5 \times 10^{10}$.

These figures may be compared with the worst uncontained accident in a 500 MW(th) reactor [6] which leads to a potential economic loss of $\$7.2 \times 10^9$.

Detailed examination of accidents [7–10] which have occured in the nuclear industry reveals that, though infrequent, they occurred mainly in nuclear chemical plants. Also, the probability of incidents in which radioactive materials are liable to be dispersed in an uncontrolled manner is greater in nuclear chemical installations than in nuclear reactors, for the following fundamental reasons.

While, in nuclear reactors, the fission products are enclosed in several sealed containments which prevent their dispersal, i.e. the nuclear fuel cladding, the reactor core vessel and the containment of the building itself, in nuclear chemical plants the nature of the operations taking place does not permit the use of such efficient enclosures.

In certain cases, the specific activity of radioactive materials processed is larger than that of the fuel containing the fission products in nuclear reactors. The specific activity of ^{90}Sr in a radioisotope source processing plant, for example, is 144 curies/g. This may be compared with the value of 6.5×10^{-3} curies/g fuel for the specific activity of ^{90}Sr in a 500 MW(th) reactor containing about 46 tons UO_2, after 180 days, operation [6]. The total specific activity of fission products 24 h after shut-down is also extremely small (9 curies/g fuel) as compared to the ^{90}Sr source specific activity.

In nuclear reactors, the operations carried out are definite and known from the outset. This is in contrast to nuclear chemical plants and laboratories, where operations are more diverse and in certain cases also difficult to define exactly at the outset. The type of

operations which are liable to give rise to uncontrolled criticality are also greater in the nuclear chemical industry.

The presence in these industries of inflammable materials which can lead to fires, and of chemicals which can lead to uncontrolled chemical reactions and even explosions, also increases the probability of accidents occurring.

Because of the potential hazards involved, the measure of attention devoted to avoiding the transformation of a potential hazard into an active danger should be certainly not less than in the planning of nuclear reactors. This is recognised, apart from other measures, by carrying out an evaluation study which stresses the apprehension of these hazards and the precautionary measures required to avoid or minimize them.

We present in this paper an outline and discussion of the particular points which we consider should be taken into account in planning the safety precautions of a nuclear chemical establishment and which should be considered in a report evaluating the hazards in such an installation. Because of the discrete nature of operations carried out, this guide cannot aspire to be comprehensive; however, details which are relevant to such plants are outlined.

SYNOPSIS OF THE HAZARDS EVALUATION REPORT OF A NUCLEAR CHEMICAL PLANT

Abstract

A summarised description of the site and plant and a synopsis of features (positive and negative) affecting the safety of the plant. A short description of the processes involved in the plant, incidents which could possibly occur and relevant safety precautions, and an appraisal of the safety of the enterprise with regard to the surroundings of the site on which the plant is erected.

Introduction

The purpose of the report. A brief description of the objectives of the plant, the history of its design, and the agencies responsible for its planning, construction and running-in. The experience and training of the responsible personnel.

Description of the site

Description of the characteristics of the site (e.g. likelihood of earthquakes or floods) which are liable to cause a hazard or to influence the safety of the plant by adding to or alleviating the results of an accident on the surroundings.

Meteorology

A general review of the seasonal climatic conditions in the area. The stations taking measurements for meteorological data of the site, the frequency of measurements, the distance and direction relative to the plant of points of measurement and applicability of information taken at off-site stations. Data supplied by the measurements: direction, speed and frequency of winds (at different heights) and their daily, monthly and yearly distribution. Average temperature, maxima and minima, daily and seasonal variations. Stability and atmospheric diffusion coefficients. Frequency and duration of laps and inversion conditions, monthly and yearly summary of the days of marked inversion and frequency of fumigation and looping conditions. Direction and speed of the winds during lapse, inversion, fumigation, and looping conditions.

Frequency of storms or other severe climatic conditions in the area, and data on winds (direction, speed, frequency) under these conditions. Summary of monthly and annual quantities of rainfall or snow and the number of rainy or snowy days. Direction, speed, and frequency of winds during rainfall or snow.

Dust load of the atmosphere.

A discussion of the meteorological information as it affects conditions at the site with regard to routine and accidental release of radioactive material.

Geology

Geological definition of the site accompanied by a cross-sectional geological map. Description of the drills and test areas on the site and surrounding it. Physical and chemical characteristics of the soil (porosity, ion exchange, drainage, permeability, corrosion). Discussion of the suitability of the geologic formation for supporting the structures of the plant and retaining radioactive materials released during an accident.

Hydrology

Description of the sources of surface and underground waters. Water level pattern. The direction and rate of flow of the water sources (linear and volumetric). Flow pattern prevailing and seasonal and annual changes. Absorption and drainage characteristics of rain water.

Exploitation of the water in the area (characteristics of wells) and evaluation of changes in the water flow pattern due to its present and future exploitation.

Frequency and strength of floods in the area. Tidal effects.

Biological constituents of the water supplies.

Topography

A topographical chart indicating position of the plant relative to other installations on the site. Heights of the hill and mountain ranges, and depths of the valleys in the area. Topographical changes which may occur due to the erection of the plant and development of the surroundings.

Seismology

Seismic survey and history of the area indicating the strength, severity and frequency of earthquakes which have taken place, and dates of occurrence. The presumed maximum strength of future earthquakes. Sensitivity of geological strata to seismic influence. Location of active geological faults in the surroundings. The seismic coefficient for designing the construction of buildings.

Distribution of population

Population centres and the distribution of population within the area liable to be influenced by an incident at the plant (indicating distance and direction). Future changes to be expected.

Radiological sensitivity

Description of agricultural and industrial activity in the area (indicating distances, financial investment, and the number of workers employed). Holiday resorts. Occupations of the inhabitants living in the surrounding area. Description of transport facilities (roads, railways) and restrictions on movement. The exclusion radius of the enterprise.

Description of the facility

Detailed description of the buildings of the facility, including the main and ancillary installations, from the standpoint of design and construction materials. Sufficient information has to be given to characterize the facility.

Objective of the operations

The object of the work and a general description of the operation stages.

Inter-functional arrangement of buildings and laboratories

Schematic diagram of the buildings on the site. Distance of the plant from other buildings and services (e.g. fire station). Height of the buildings and the topographical height of the site on which they are situated. Schematic diagrams of the laboratories and principal buildings, indicating the principal facilities (glove boxes, hot cells, etc.). Classification of buildings and laboratories in regard to the amount and class of radioactive materials in use.

Design

Description of design of the building, thickness and materials of construction of the walls, roof and floor. Position and dimensions of doors and windows (indicate double doors with airlocks) and the degree to which they are leakproof, direction of opening, and materials of construction. Interlock and bypass provisions for airlocks. Emergency escape doors. Detailed description of each of the rooms of the building; size, thickness of the walls, materials of construction, paintwork, degree of resistance to fire and to radioactive contamination.

Design of the facilities in which the various operations are carried out (glove boxes, hot cells, fume hoods, etc.). Dimensions of facilities, materials of construction, and degree to which they are leakproof. Indication of method of introduction and removal of materials into and from the various facilities (including description of change of gloves in the glove boxes). Resistance of specific equipment to pressure and fire and details of the safety limits of the various equipment operating under extreme conditions (pressure, temperature).

Indication of penetrations, spray systems or other means for pressure reduction and of airlocks and interlocks, in the containment facilities. Estimated corrosion and erosion rate of the constructional materials of the different facilities. Corrosion control measures.

Description of ancillary installations

Ventilation

General description of the system, including joints and welds in the ventilation ducts and their tightness to leaks. Number of air changes per unit time in the different facilities. Air pressure conditions prevailing in various regions and facilities. Linear air flow rate in the various facilities. Description of testing of the ventilation system and the degree of leakproofness.

Filtration

Types and materials of filters and their location in the ventilation system, their fire resistance, and efficiency with respect to different materials. Method and frequency of testing the efficiency of filters, and their accessibility as regards replacement.

Water supply

Description of the potable process and cooling water systems.

Electricity

Description of the electric installation, indicating cables within or on the surface of walls.

Supply and other service installations

Description of other supply systems (combustible gas and other gases). Description of special ancillary installations of the works (compressed air supply, etc.).

Monitoring and sampling of air and cooling water

Description and location of the permanent installation for monitoring activity of the air and cooling water inside and outside the plant, indicating the

sensitivity and the existence of double monitoring. Instruments available to the Health Physics Department. Periodical testing and calibration of instruments.

Instrumentation

Description of the various instruments for monitoring pressure, flow rates, temperature, humidity, the fluid levels in closed containers, and of instruments for the detection of fire and explosion, etc., indicating response time and sensitivity, and the extent of double monitoring.

Description of the periodic testing of the instruments.

Emergency supplies

Description of the emergency supplies of water and electricity, their mechanism of activation and the equipment and facilities to which they are connected, indicating the delay at which they can be put into operation. Quantity of reserve water and power of electricity supply system for emergency conditions.

Emergency lighting system.

Description of the warning, alarm, and fire extinguisher installation.

Description of the emergency ventilation system. Existence of emergency reserve equipment to replace installations which may be damaged by an incident (glove boxes, containers, etc.).

Description of the decontamination room, including materials and facilities.

Description of the processes involved

A detailed description of the processes, methods of production (accompanied by a quantitative flowsheet) and types of experiments, underlining the critical points.

The processes (chemical, metallurgical, and others)

Description of production method or experiments, indicating all the materials in use, their properties (e.g. pyrophoricity) and the various parameters which affect reactions (e.g. temperatures and pressures prevailing at the various stages). Description of the equipment in which the various operations are carried out, indicating the geometry and the distances between them, for fissile materials.

Description of the atmosphere in which the operations are carried out (air, rare gas, or other gas) and the extent of its dryness and purity.

Description of the method of transfer of materials from one installation to another.

Description of the control system, interlocks, control action (including a block diagram), indicating fail safeness and an over-all notion of what actions take place under specific operating conditions. Means available for communication between the control room and other parts of the plant. Audible and visible alarms throughout the plant and in the control room.

The maximum quantity of material likely to be present in each facility

A quantitative flowsheet of materials involved in the processes, indicating physical state and concentration. The maximum quantity of radioactive or poisonous materials to be expected in each facility and installation, indicating their identity, their physical state and concentration, their chemical properties, and their type of radiation and energy. Specification of procedure for keeping the inventory record of the materials.

The estimated maximum energy and pressure in every facility

Reactions

Calculation of the energy liable to be released in every reaction, indicating reactions which may occur even with low probability. Change of pressure (as a function of time) expected in the installation and equipment in which the reactions take place.

Escape of gas

Gaseous materials liable to be released in the production processes or in experiments, or uncontrolled release from storage containers.

Inspection and testing of the installations

Description of the testing and inspection procedure for the installations, and its frequency. Pre-operational checks.

Maintenance

Description of maintenance procedures and facilities, lifting and handling equipment and decontamination facilities.

Storage of materials

Radioactive materials

Description of the materials stored (physical state, quantity, radiation) and the storage container. The type of ventilation in the storage vault. Cooling and shielding provisions. Alarm system in the storage vault, and safety precautions in the event of mishap.

Fissile materials

Description of storage materials (physical state, quantity). Description of storage containers, their geometry, and the distance between them. Concentration of solution (in the case of liquid storage); description of shielding, cooling, ventilation and alarm systems in storage vaults, and safety precautions against incidents.

Inflammable material

Description of materials (quantity, physical state). Description of storage containers, ventilation, cooling and alarm systems, and emergency safety precautions.

Poisonous material

Description of materials (quantity, physical state), manner of storage, ventilation and alarm systems, and safety precautions against incidents.

Transport and carriage

Description of the means of transfer and conveyance of various materials within and without the facility, indicating containers used and safety precautions.

Waste disposal

Description of the collection system and methods of disposal of gaseous, liquid and solid waste, the maximum anticipated quantity of waste, its composition (isotopic identity) and specific activity. The concentration of solids in liquid waste. Shielding of the waste disposal system.

Description of waste disposal installations. Gaseous waste—the chimney and its dimensions, filters, and means of adsorpion. Liquid wastes—the design of sewage systems, description of drains and syphons in the system. Capacity and materials of construction of the containers of the sewage system for liquid wastes at different levels of radioactivity. Life expectancy of containers in service. Treatment of liquid wastes (concentration, vaporisation, discharge to sea or rivers). Solid waste—description of treatment and place of storage.

Provisions for emergency waste storage.

Shielding

Biological shielding

Description of shielded facilities, shielding materials and dimensions. Susceptibility to radiation damage. Radiation dose rates outside the shield.

Blast and missiles shielding

Location and materials of construction of the shield. Resistance of facilities and buildings to pressure, explosion, and missiles.

Containment and safety precautions

Policy adopted for safety devices and containment, including the various sealed containers for the materials (containers, fume hoods, glove boxes, hot cells, safes, and other special containers) and the extent to which they are leakproof.

Organization and responsibilities of personnel

Organization of the facility. Education, qualifications and training of the personnel, their functions and responsibilities. Safety and advisory councils, their functions and responsibilities, and extent of their independence of the operating team. Routine operating and safety instructions. Special instructions for use of different installations. Adherence to written instructions.

Accident analysis

Analysis of possible accidents arising from routine and abnormal operation, technical mishaps, "acts of God" (earthquakes, floods) and other causes (sabotage, war). Description of the initiating events, of the manner of detecting the incidents and of the safety precautions adopted (e.g. alarm, closure of ventilation), indicating whether carried out automatically or manually, and the delay at which the emergency safeguards are put into operation.

Explosion

Analysis of events liable to lead to an explosion arising from errors in carrying out the production process or the experiments, or arising from technical failures. Evaluation of the maximum energy released by the explosion and the pressure produced in the installation. Safety precautions to avoid or reduce the magnitude of incidents and their consequences.

Criticality accidents

Estimation of evolution of a chain reaction in fissionable material due to increase of mass of material, change in shape of the vessel, inadvertent introduction of a reflector, uncontrolled addition of moderators, formation of precipitates or slurries, or unusual spacing between containers, or other conditions resulting from human error. Description of safety precautions to avoid or reduce the extent of incidents. Indication of presence of neutron poison, as a safety precaution. Method of detection and warning and instructions to personnel.

Dispersal of radioactive or poisonous material

Analysis of the events leading to the possibility of dispersal of radioactive or poisonous material, as in the transfer of material from one installation to another, explosion, leakage from vessels and other enclosure facilities, or change of pressure in the installations. Extent of the dispersal. Methods of detecting contamination. Safety precautions and instructions to personnel.

Nuclear reactions

Analysis of accidents arising from nuclear reactions, e.g., the reaction (α, n) between alpha emitters and light elements.

Failure in the ventilation system

Analysis of failures liable to occur in the ventilation system, filters and chimneys, and of the consequences.

Detection and warning systems and precautionary measures to avoid or reduce the effect of accidents.

Failure of electrical system

Analysis of the situation expected in the event of electrical failure as a result of general or partial breakdown of the supply network (failure of the ventilation, increase of pressure, disturbances in the production processes or experiments etc.). Safety precautions and emergency system.

Instrument failure

Indication of fail safeness of the different instruments or analysis of consequences of failure.

Failure in the water system

Analysis of incidents arising from stoppage of flow of water or uncontrolled leakage outward from or inward to the operating installation. Methods of detection. Safety precautions and emergency installation.

Fire

Accident analysis of local or general fire outbreak and the results expected. Warning instruments. Methods and materials for fire extinction.

Earthquake

Analysis of incidents which are liable to be caused by earthquakes (breaking of pipes, structural failure, cracks in the construction, damage to the containment enclosures, etc.).

War and sabotage

Evaluation of the damage which can be caused to the plant by sabotage and war action.

Consequences of incidents

Summary of the consequences to be expected from the initiating events described in the previous section, from the point of view of release of energy, build-up of excessive pressure in the various installations, missile formation and impact, and dispersal of radioactive or poisonous material. Description of the accident which from the point of view of release of radioactive or poisonous materials is considered the maximum credible accident.

Hazards analysis

Over-all appraisal of the hazards inherent in the enterprise from the point of view of likeliness of release of release of radioactive or poisonous materials. Detailed description (indicating methods of assessment) of the exposure of workers and the environmental population to internal and external contamination or radiation.

Evaluation of the damage to land expected in the environmental area by contamination resulting from deposition of the materials.

Maximum permissible levels of radiation and contamination

The tolerable levels of contamination and radiation doses to workers and the population under normal conditions of operation and under emergency conditions after an incident. The tolerable levels of contamination to farming lands.

Hazards in normal operation

The characteristics of release

The source of the materials liberated from the plant to the surroundings (controlled release, uncontrolled leakages, cooling water, effluent streams, etc.). Type and identity of materials released, their chemical composition, physical state, quantity, their concentration and activity, the size of the particles, volatility, half-life of radioisotopes, adsorption characteristics, velocity of movement through the ground, etc.

Transport of released materials in the sourroundings

Ground level concentration of released materials as a function of distance from the installation, under various meteorological conditions (lapse, inversion, looping, and fumigation).

Concentration of materials in ground water, rivers and lakes in the environment of the installation.

Evaluation of radiation doses to the population

External radiation from radioactive cloud as a function of distance from the plant.

Internal radiation doses from inhalation of contaminated air and ingestion of contaminated water or contaminated food as a function of distance from the plant.

Indication of critical body organs affected.

External radiation dose from material deposited on land under dry weather, rain-out and wash-out conditions.

Monitoring in the surroundings

Description of monitoring equipment in the environment of the plant and its sensitivity. Indication of frequency of measurements.

Emergency release of materials during incidents

Estimation of leakage

Estimation of leakage of material, under accident conditions, from the different installation facilities to the building and thence to the environment.

Source, quantity and identity of materials released

The quantity and identity of materials released and their characteristics (physical and chemical state, size of particles, volatility, adsorption characteristics, activity and half-life of radioisotopes, etc.).

The nature of release

Description of cases in which the building remains undamaged and there are only leakages (continuous release), and the case where the building is ruptured, or when the doors remain open for a limited period (momentary puff release).

Transport of released materials in the surroundings

Ground level concentration of released materials as a function of distance from the installation, under various meterological conditions. Height of cloud release and its dimensions as a function of distance from the plant and atmospheric conditions. The manner in which the released material reaches the ground water, rivers, or lakes after an incident. Concentration of isotopes in the water at different distances from the plant.

Computation of the radiation dose rate and integrated doses

Direct radiation from the building: the radiation dose rate received outside the installation as a function of distance, in the event of an accident in which the building remains sealed and the radioactive material is dispersed within it.

The external radiation doses from the radioactive cloud: the distance at which the maximum concentration occurs under different atmospheric conditions, and radiation dose as a function of distance from the plant. The internal radiation doses from inhalation of contaminated air or ingestion of contaminated food, as function of distance from the plant. Details of the critical body organs affected.

The external radiation dose rate from material deposited on land. Computation of the radiation doses from deposited material under dry weather, rain-out and wash-out conditions.

Computation of the isodoses of integrated exposure as a function of delay after release and distance from the plant.

Area contamination

Estimation of levels of contamination of the environmental area resulting from deposition of radioactive material under different atmospheric conditions, and expenses liable to be caused by the contamination and prohibition of use of land and crops.

Emergency procedures

Description of emergency procedure and equipment. The location of the monitoring and emergency equipment on site and off site. Decontamination procedure.

Indication of the facilities which will continue to operate under emergency conditions.

On-site and off-site evacuation procedure.

Definition of personnel duties.

Evaluation of the hazards of the plant

Summary of the conclusions drawn from the hazards evaluation study of normal operation and accident conditions in the installation.

Evaluation of the number of persons liable to be injured and the extent of the injury.

Estimation of the necessity of evacuating the environmental population after an accident.

The damage to be expected to agricultural and industrial activities in the surroundings and evaluation of the necessity of imposing restrictions on the environment by establishing an exclusion and restricted area radius, as a result of the potential hazards involved in the operation of the nuclear chemical installation.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/508 Israël

Guide pour l'évaluation des risques présentés par une installation de chimie nucléaire

par J. Tadmor et H. Galron

L'expérience acquise dans l'exploitation des installations nucléaires a montré que — malgré leurs dangers potentiels — la sécurité y était une des meilleures dans le secteur industriel. Ce résultat est la récompense des efforts faits pour discerner les risques présentés par les installations de chimie nucléaire, éviter les défaillances et les accidents et, le cas échéant, réduire au minimum les suites possibles des accidents.

L'étude de l'évaluation des risques faite avant la construction et la mise en service d'une usine nucléaire — étude adoptée comme procédé standard — constitue un exemple des efforts spéciaux faits dans le domaine de la sûreté nucléaire.

Le grand nombre de rapports publiés qui traitent de l'évaluation des risques des réacteurs nucléaires, de même que les guides qui servent à les formuler, donnent une idée des problèmes de sécurité que pose leur exploitation. Par contre, dans le domaine des installations de chimie nucléaire, relativement peu de rapports, ont paru jusqu'à présent sur l'évaluation des risques.

Les auteurs présentent une étude sur les risques pour l'environnement créés par l'exploitation d'une installation typique de chimie nucléaire; cette étude est basée sur le concept de l'accident « maximal croyable » et elle fait une comparaison avec les risques pour l'environnement que comporte le fonctionnement d'un réacteur nucléaire.

Le mémoire discute les données générales nécessaires pour l'étude d'une évaluation des risques d'une installation de chimie nucléaire, risques qui devraient être pris en considération lors de la planification des mesures de sécurité pour ces usines. Parmi les sujets dont la connaissance est importante pour l'évaluation des risques figurent:

Evaluation de l'emplacement. Matériaux de construction et résistance de l'installation au feu, à la pression et aux ondes de choc. Installations auxiliaires (ventilation, filtration, système de contrôle, dispositifs d'urgence, etc.). Etude de divers processus (chimique, métallurgique), conditions de fonctionnement (température, pression, etc.) et des matériaux en jeu et de leurs propriétés (par exemple, radioactivité, pyrophoricité). Système de commande. Entretien et fonctionnement, stockage, transport, élimination des déchets. Blindage et enveloppes de sécurité. Organisation et attributions du personnel. Accidents possibles (feu, explosion, criticité, tremblement de terre, etc.) et leurs conséquences. Evaluation des risques au cours d'une exploitation normale et continue ainsi qu'à la suite des accidents. Mesures d'urgence.

Une place spéciale est réservée à diverses caractéristiques des usines de chimie nucléaire qui les différencient des réacteurs au point de vue de la sûreté: activité nucléaire volumique élevée des matières, présence de matières inflammables et explosives, variété des opérations, et impossibilité d'assurer dans les installations le même degré de confinement que pour les réacteurs nucléaires.

A/508 Израиль

Инструкция по оценке опасностей работы на ядерно-химической установке

Дж. Тадмор, Х. Гальрон

Опыт эксплуатации показал, что, несмотря на потенциальную опасность ядерных установок, их безопасность обеспечена лучше, чем безопасность установок, используемых в других отраслях промышленности. Это явилось результатом проведения исследований, направленных на изучение опасностей работы на химических установках по переработке облученного горючего, устранения замеченных недостатков и причин аварий, а также стремления свести к минимуму возможные последствия аварий.

Изучение опасностей ядерной установки, как правило, проводимое до ее стоительства и ввода в эксплуатацию, является примером достигнутых успехов в области ядерной безопасности.

Большое количество опубликованных работ по оценке опасностей ядерных реакторов, а также руководств, в которых перечисляются опасности, связанные с их эксплуатацией, освещает общие вопросы безопасности, касающиеся эксплуатации ядерных реакторов. Вместе с тем до последнего времени по оценке опасностей ядерно-химических установок, было опубликовано относительно мало работ.

В настоящем докладе рассматриваются опасности для окружающей среды, имеющие место при эксплуатации типичных ядерно-химических установок. Оценка опасностей производится исходя из «предельно возможной аварии». Сравниваются опасности для окружающей среды, обусловленные химическими установками по переработке облученного горючего и ядерными реакторами.

В докладе содержится общая информация, необходимая для проведения оценки опасностей химической установки по переработке облученного горючего, которую следует иметь в виду при планировании мер по обеспечению безопасности работы на этих установках. Рассматривается как радиационная, так и химическая безопасность. При оценке опасностей наиболее важна информация по следующим вопросам: место размещения; строительные материалы и противопожарная безопасность установки; возможное давление и ударные волны; вспомогательные установки (вентиляция, фильтрация, система контроля, аварийные средства и пр.); результаты изучения различных процессов (химического, металлургического), условий эксплуатации (температура, хранение, транспортировка, удаление отходов, ответственность); сведения о возможных авариях (огонь, взрыв, надкритичность, землетрясение и пр.) и их последствиях; сведения об опасностях в процессе нормальной непрерывной эксплуатации и при авариях, о срочных мерах.

Специальное место в докладе отводится различным характеристикам ядерно-химических установок, которые отличают их от реакторов с точки зрения безопасности: высокая объемная удельная активность материалов, разнообразие операций, невозможность обеспечения установок защитой на том же уровне, что и в ядерных реакторах.

A/508 Israel

Guía para estimar el riesgo de una instalación químico-nuclear

por J. Tadmor y H. Galron

La experiencia adquirida en la explotación de las instalaciones nucleares demuestra que, a pesar de los riesgos potenciales de esta industria, su expediente de seguridad figura entre los mejores de los del campo industrial. Se ha conseguido este resultado realizando esfuerzos sin parangón para identificar los riesgos de las instalaciones nucleares, para evitar la producción de averías y accidentes y para reducir al mínimo los efectos posibles de un accidente, si éste llegase a ocurrir.

Los estudios de evaluación de riesgos realizados antes de la construcción y de la explotación de una instalación nuclear, estudios adoptados como procedimiento normal, son un ejemplo de los esfuerzos especiales llevados a cabo en el campo de la seguridad nuclear.

El gran número de publicaciones sobre la estimación de los riesgos de los reactores nucleares, así como las guías para formularlos, permiten el reconocimiento de los problemas generales de seguridad encontrados en la explotación de los mismos. Sin embargo, se han publicado relativamente pocos informes sobre las instalaciones químicos-nucleares.

Se analizan los riesgos exteriores asociados con la explotación de una instalación químico-nuclear representativa; los añálisis se basan en el concepto del « máximo accidente probable » y los resultados se comparan con los riesgos exteriores de los reactores nucleares.

En esta memoria se discute la información general que se necesita para evaluar los riesgos de una instalación químico-nuclear; tal información debe tenerse en cuenta al planear las precauciones de seguridad para estas instalaciones. Se consideran dos tipos de seguridad: seguridad nuclear y seguridad química. Entre los temas sobre los que se necesita información para el estudio de la evaluación de riesgos se cuentan los siguientes:

Evaluación del emplazamiento. Materiales de construcción y resistencia de la instalación al fuego, las ondas de presión y de choque. Instalaciones auxiliares (ventilación, filtración, vigilancia radiológica, sistemas de emergencia, etc.). Estudio de los diferentes procesos (químicos, metalúrgicos), de las condiciones de funcionamiento (temperatura, presión, etc.) y de los materiales utilizados y de sus propiedades (por ejemplo, radiactividad, piroforicidad). Sistema de control. Entretenimiento y explotación. Almacenamiento, transporte y eliminación de residuos. Blindaje y contención. Organización del personal y sus responsabilidades. Accidentes previsibles (fuego, explosión, criticidad, terremotos, etc.) y sus consecuencias. Estimación de los riesgos durante la explotación normal y continua y después de los accidentes. Planes de emergencia.

Se da importancia especial a aquellas características de las instalaciones químico-nucleares que las diferencian de los reactores nucleares desde el punto de vista de la seguridad, tales como la elevada actividad específica de los materiales, la presencia de sustancias inflamables y explosivas, la gran variedad de operaciones llevadas a cabo y la imposibilidad de construir dichas instalaciones con el mismo grado de contención que los reactores nucleares.

Études de criticité

par D. Breton, P. Lecorche, C. Clouet d'Orval *

Les études de criticité entreprises comportent d'une part des expériences sur des solutions de matières fissiles, d'autre part la mise au point de méthodes de calcul appliquées à ces expériences.

Les expériences portent sur des cylindres, des réservoirs annulaires et des plaques (isolés ou en interaction), dans diverses conditions de concentration et de réflexion.

Parallèlement, des méthodes de calcul (diffusion, transport, Monte-Carlo) ont été mises au point et leurs résultats comparés à ceux des expériences. Elles permettent en outre de traiter les problèmes industriels posés.

Quelques exemples montrant comment ont été résolus certains de ces problèmes mettent en évidence les critères de sécurité employés couramment.

EXPÉRIENCES FAITES A SACLAY APPAREILLAGE ET RÉSULTATS

Les études critiques entreprises à Saclay portent essentiellement sur des récipients cylindriques contenant des solutions fissiles à concentration variable, soumises à diverses conditions de réflexion et éventuellement d'interaction avec d'autres unités fissiles. Un ensemble de résultats expérimentaux, obtenus pour divers diamètres de cuves, est présenté, ainsi que des résultats de calculs portant sur ces différents cas.

Appareillage expérimental ALECTO (^{239}Pu, ^{235}U, ^{233}U) [1, 2]

Les deux expériences ALECTO sont situées à Saclay dans un hall étanche de 30 × 15 m² qui abrite également la pile PROSERPINE, première expérience critique homogène au plutonium ou à l'uranium 235 [6].

ALECTO I concerne le plutonium, ALECTO II l'uranium enrichi à 90 %. Dans ces ensembles, on peut rendre critiques des solutions fissiles contenues dans des cuves cylindriques, de diamètres variables entre 20 et 50 cm, de hauteur allant jusqu'à 100 cm, dans différentes dispositions de réflexion et d'interaction. Chaque installation se compose essentiellement d'une cabine étanche, contenant l'assemblage critique, et d'une cabine de stockage.

La cabine pile contient la cuve en essai, d'épaisseur 3 mm, surmontée d'une boîte à gants, dans laquelle se trouvent suspendues deux barres de sécurité formées de sandwiches cadmium-acier inoxydable, une barre de commande et une pointe de mesure de niveau. La cuve peut être désolidarisée de la boîte à gants supérieure qui peut alors, en prenant toutes précautions pour éviter la contamination, être adaptée à une cuve d'un diamètre différent. La cuve peut être entourée d'un réflecteur infini d'eau ou d'un réflecteur solide (béton, bois, paraffine) ou d'un réflecteur composite pouvant contenir du cadmium ou de l'acier, ou encore de bouteilles entrant en interaction avec la cuve. Cuve et boîte à gants supérieure sont démontables et peuvent être séparées du tube d'amenée de la solution fissile provenant du stockage.

Les stockages contenus dans des cabines séparées sont de géométrie sûre. Ils sont essentiellement composés de réservoirs annulaires, réfléchis intérieurement par de la paraffine, une feuille de cadmium étant intercalée entre le réservoir et la paraffine. L'épaisseur du cylindre annulaire contenant la solution est de 40 mm pour le plutonium, 50 mm pour l'uranium enrichi. Une série de bouteilles de diamètre 100 mm, écartées de 600 mm d'axe en axe, permet de compléter ce stockage et d'effectuer les transferts.

L'appareillage est caractérisé essentiellement par une grande souplesse de manipulation, compte tenu des risques de contamination. Une détection des fuites de solution est possible en de nombreux points. La montée des solutions est effectuée en deux temps, par deux pompes différentes. Au début de la montée, la variation de réactivité est au maximum de 45.10^{-5} dK/K par seconde; à partir d'un niveau contrôlé par une pointe fixe, elle ne peut être supérieure à 14.10^{-5} dK/K par seconde.

Résultats expérimentaux **

Des cuves de diamètre variable ont été expérimentées au moyen de ces assemblages critiques. Outre la cuve ($\varnothing$ = 330 mm) à fond bombé [3], on peut citer les

* Commissariat à l'énergie atomique. Avec la collaboration de R. Caizergues, J. Moreau et G. Sollier.

** Des résultats récents, concernant en particulier l'uranium 235, sont présentés sur la figure 9.

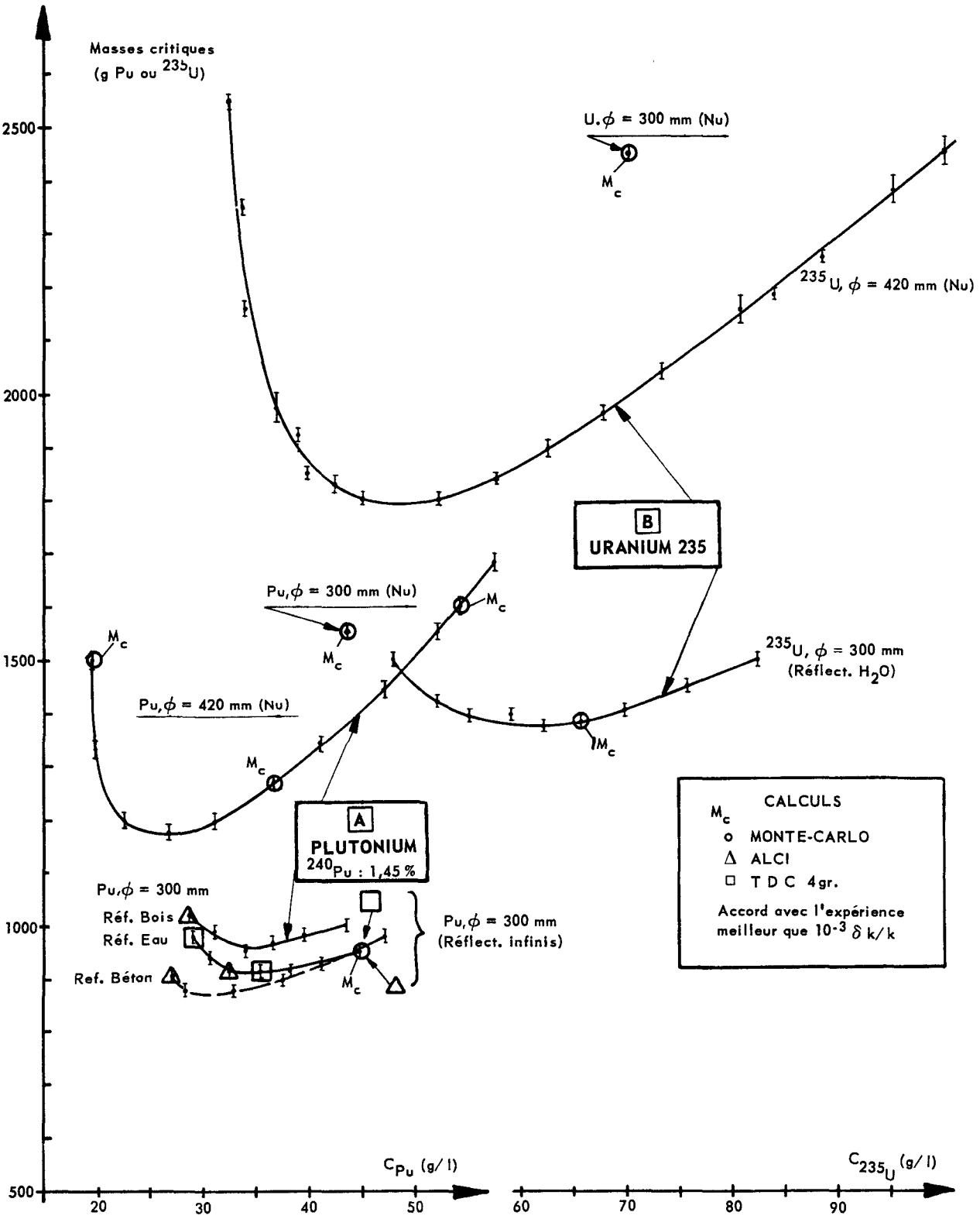


Figure 1. Masses critiques. A: plutonium; B: uranium 235 enrichi à 90%

cuves ⌀ = 300 mm [4] et ⌀ = 420 mm [5] dont nous présenterons quelques résultats. Il est à signaler que dans les deux cas, plutonium et uranium enrichi, la géométrie était absolument identique, ainsi que, le cas échéant, les conditions de réflexion.

Les solutions fissiles contiennent un sel dissous, $(NO_3)_4$ Pu ou $(NO_3)_2UO_2$ dans l'eau légère, en présence d'une acidité libre, toujours très voisine de $2N$. Les mesures de concentration, d'acidité, de densité, sont données dans les références.

Masses critiques *

Les résultats peuvent être groupés, pour le plutonium et l'uranium, selon les dimensions de la cuve:

a) Cuve de 420 mm, sans réflecteur.

b) Cuve de 300 mm, nue ou avec divers réflecteurs latéraux.

L'ensemble des courbes plutonium et celui des courbes uranium sont représentés sur la figure 1 A et B. Il est à remarquer que, dans notre gamme de concentrations, le rapport des masses critiques minimal du plutonium et de l'uranium est, pour diverses géométries nues ou réfléchies, voisin de 5/8.

* Des résultats récents, concernant en particulier l'uranium 235, sont présentés sur la figure 9.

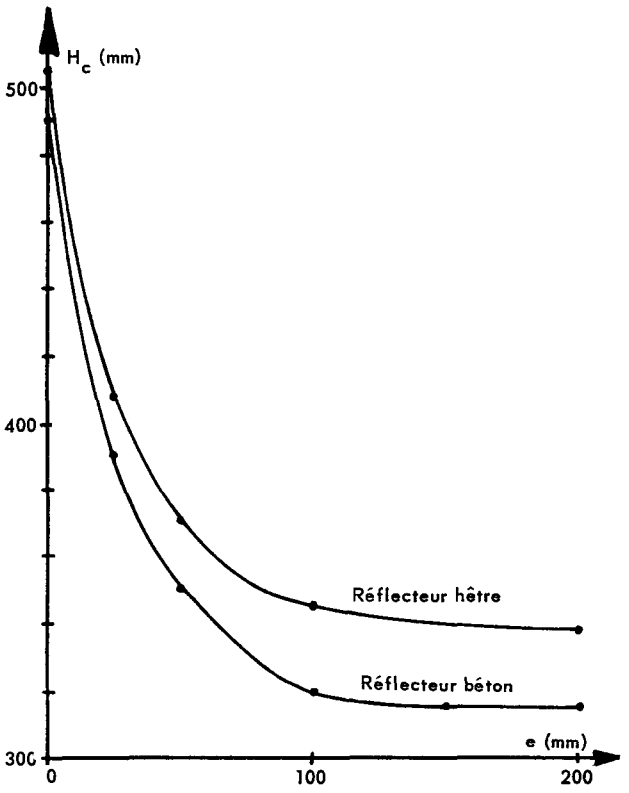


Figure 2. Effet de l'épaisseur du réflecteur. Solution plutonium, cuve ⌀ 300

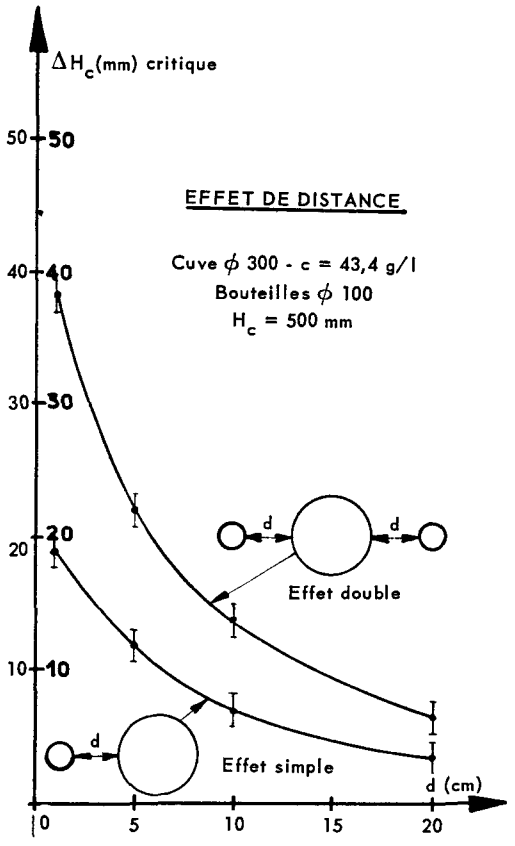


Figure 3. Effet de distance

Cuve ⌀ = 43,4 g/l; bouteilles: ⌀ 100; H_c = 500 mm

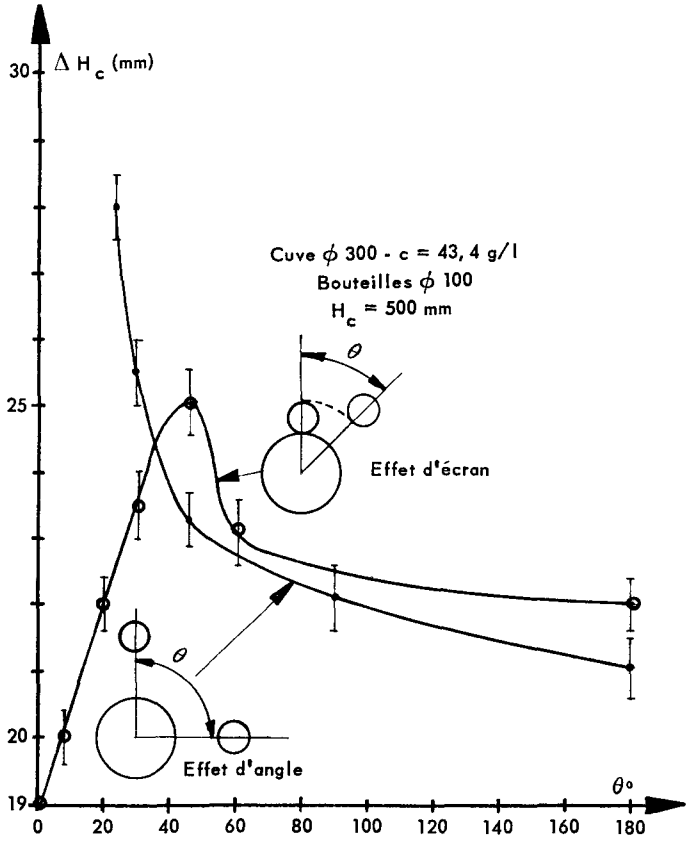


Figure 4. Cuve ⌀ 300 — c = 43,4 g/l; bouteilles: ⌀ 100; H_c = 500 mm

Par ailleurs, dans le cas du plutonium (cuve ∅ = 300 mm), le réflecteur de béton apporte plus de réactivité que le réflecteur d'eau, qui lui-même en apporte plus que le réflecteur de bois.

On a indiqué sur les courbes les résultats de calculs effectués sur ces cas expérimentaux, et dont on parle plus loin.

Effets divers

a) *Effets de réflexion, de la cuve, d'un absorbant*
Cuve ∅ = 300 mm (plutonium)

La masse critique en fonction de l'épaisseur des réflecteurs (fig. 2) a été déterminée sur la cuve de ∅ = 300 mm, dans le cas du plutonium, pour le béton et le bois. Un réflecteur d'épaisseur 150 mm, a été dans les deux cas trouvé équivalent au réflecteur d'épaisseur infinie. L'effet de l'épaisseur de la paroi pour la cuve de ∅ = 300 mm en réacteur nu, a été déterminé par addition de couches successives de 3 mm et 6 mm d'acier inoxydable. Par extrapolation à une épaisseur nulle, on a trouvé que l'effet de la cuve d'acier inoxydable, uniquement par ses surfaces latérales, correspond à une variation de 15% de la hauteur critique.

Par ailleurs, on a trouvé, à la concentration de 43,6 g/l, qu'un réflecteur infini de bois de hêtre plus une feuille de cadmium entre cœur et réflecteur était équivalent à un réflecteur d'épaisseur 4 cm. Dans le cas d'un réflecteur infini, une feuille d'acier inoxydable (e = 3 mm) n'a pas grande influence sur la taille critique. Enfin, une feuille de cadmium de 0,7 mm d'épaisseur enroulée autour d'un réacteur nu apporte une légère réactivité positive (200 pcm pour une hauteur de solution de 500 mm).

b) *Effets d'interaction — Cuve ∅ = 300 mm (uranium)*

Une étude d'interaction portant sur des couplages faibles a été entreprise dans le but de vérifier la méthode de l'angle solide. Autour de la cuve ∅ = 300 mm, on a disposé des bouteilles ∅ = 100 mm contenant des solutions fissiles. On a ainsi mis en évidence des effets liés à la concentration, lorsque la géométrie reste inchangée.

A concentration constante, on a étudié des effets liés au déplacement des bouteilles, particulièrement interaction entre bouteilles et effets d'ombre (fig. 3 et 4). On a également vérifié certaines lois concernant l'effet de réflexion, de semi-réflexion, d'un mur, d'un écran.

EXPÉRIENCES FAITES A DIJON APPAREILLAGE ET RÉSULTATS

La Station de criticité de Dijon a été décrite dans [7]. Le premier appareillage, appelé « Appareillage B » [8], monté dans la cellule 2 et la sous-cellule correspondante, est destiné à l'étude de récipients divers contenant des solutions de nitrate de plutonium. Celles-ci sont préparées et analysées dans un atelier installé dans la Station elle-même.

Description de l'appareillage B

Cet appareillage permet l'étude des conditions de criticité de récipients isolés ou en interaction, nus ou dans diverses conditions de réflexion. Il se compose d'une cuve parallélépipédique de: 3 200 × 2 100 × 1 500 mm alimentée en eau depuis un stockage placé en sous-cellule par l'intermédiaire d'une pompe centrifuge, sur le refoulement de laquelle est placée une vanne motorisée qui permet de régler le débit.

Cette cuve sert en outre de support et de plan de référence pour les récipients à essayer. Ceux-ci, dont le nombre peut atteindre 7, sont reliés par canalisations souples munies ou non de vannes, à un « arbre de Noël » placé en bout de la canalisation d'arrivée des solutions de plutonium qui sont stockées en sous-cellule dans un assemblage de cylindres de diamètre sûr. La montée des solutions est assurée par un dispositif pompe centrifuge-vanne motorisée analogue à celui utilisé pour l'eau du réflecteur. La pompe plutonium sert également à assurer par circulation en circuit fermé dans le stockage une bonne homogénéisation des solutions avant emploi. La mesure du niveau de la solution dans les récipients et de l'eau du réflecteur est faite au moyen de pointes palpeuses à mieux que 0,1 mm sur une course de 1 000 mm. Les déterminations du niveau critique s'effectuent de façon classique par approches sous-critiques éventuellement contrôlées par divergence.

Expériences réalisées à l'aide de l'appareillage B

Une première série d'expériences a porté sur l'étude d'un réservoir annulaire en acier inoxydable (épaisseur de paroi 3 mm) de diamètre extérieur 500 mm et diamètre intérieur 300 mm. Elle a été suivie d'expériences sur deux réservoirs identiques en interaction.

La cavité centrale était vide ou remplie d'eau, une feuille de cadmium de 0,7 mm étant ou non plaquée sur sa paroi interne. Le réservoir était nu extérieurement ou réfléchi à sa partie inférieure et sur les côtés par un réflecteur d'eau de niveau égal à celui de la solution.

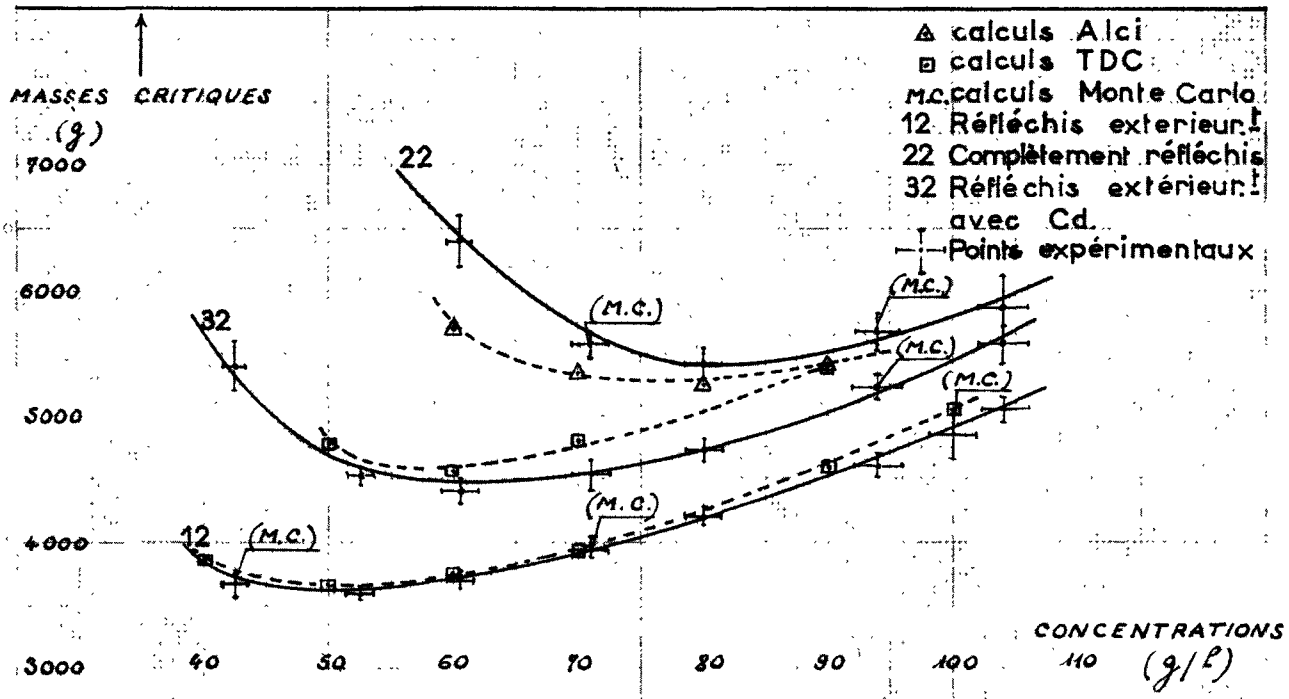
Dans le cas de deux réservoirs en interaction, l'ensemble des deux réservoirs pouvait être réfléchi en dessous et sur les côtés, l'espace entre les réservoirs étant vide.

Les expériences réalisées ensuite portent sur:

Une plaque d'épaisseur 100 mm;

Un réservoir annulaire de diamètres 500-200 mm;

Un réservoir annulaire de diamètres 500-350 mm.



CARACTÉRISTIQUES DES SOLUTIONS DE $(NO_3)_4Pu$ UTILISÉES

Concentration Pu (g/l) ± 2%	H^+ N ± 2%	NO_3^- N ± 1%	Densité mesurée à 25,8°C ± 0,1%	Teneur en Pu 240 %
104,	2,20	3,70	1,240	3,20
94,	2,15	3,59	1,222	3,20
80,	2,11	3,35	1,201	3,20
74,2	2,38	3,67	1,206	2,90
60,5	2,03	3,01	1,166	3,20
56,1	2,27	3,28	1,177	2,98
40,6	2,23	3,12	1,143	2,98

Figure 5. Cylindres annulaires isolés

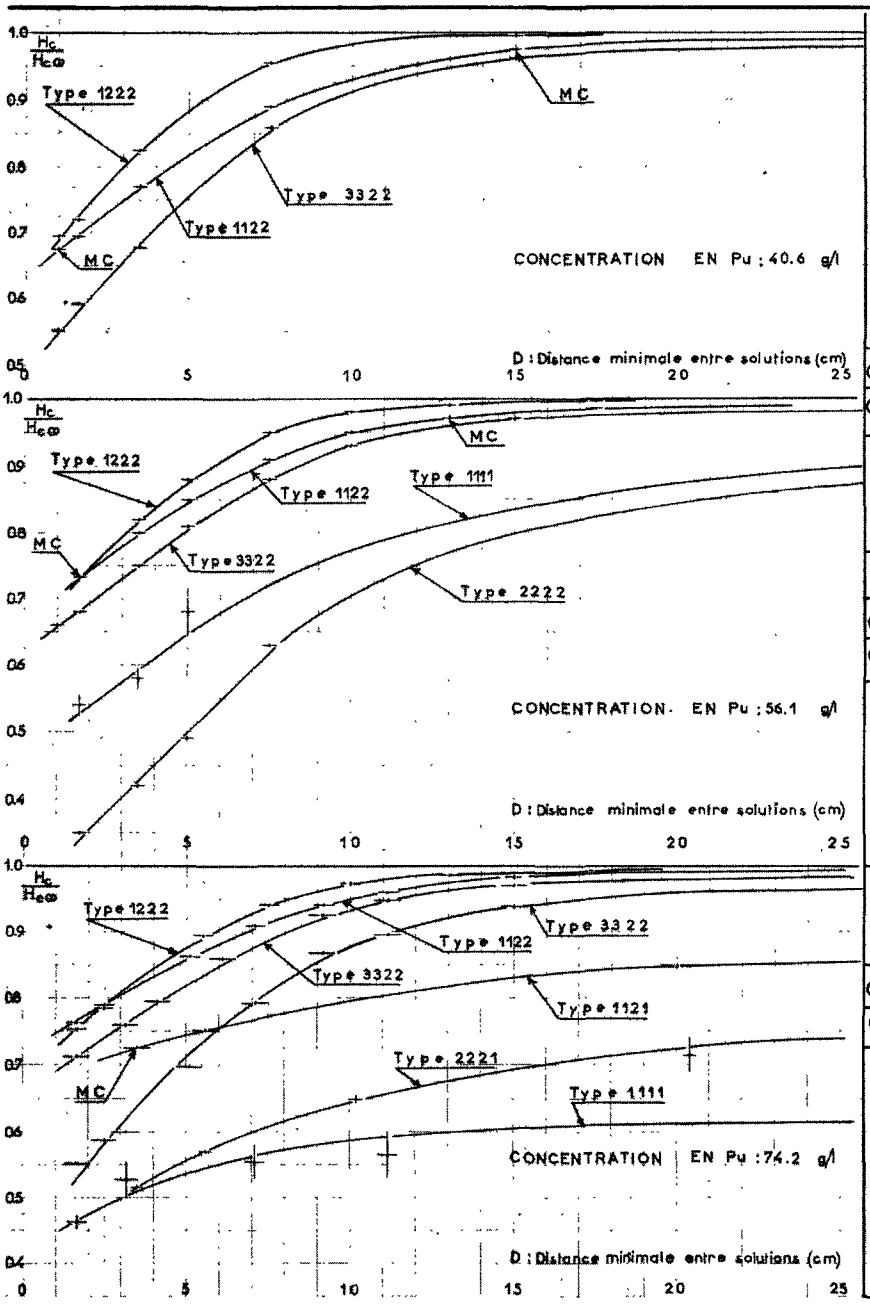
La plaque et le réservoir de 500-200, ayant été expérimentés avec des solutions d'acidité de l'ordre de 4*N*, le seront ultérieurement avec des solutions d'acidité voisine de 2*N*.

Résultats obtenus

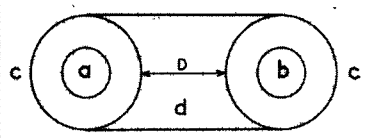
Les résultats obtenus avec les réservoirs annulaires 500-300 sont portés sur les courbes des figures 5 et 6.

Cylindre isolé

Pour une concentration donnée, le cylindre nu intérieurement est le plus réactif. Si l'on met contre la paroi interne une feuille de cadmium de 0,7 mm d'épaisseur, la configuration obtenue est plus réactive que si la cavité est pleine d'eau.



CONFIGURATION a,b,c,d.



1. AIR
2. EAU
3. AIR + FEUILLE DE CADMIUM
4. EAU + FEUILLE DE CADMIUM DE 0.7 mm PLAQUÉE SUR LA PAROI DU POT

CONCENTRATION : 40.6 g Pu par l	
Configuration	Hc ∞ (cm)
1 1 2 2	71,97
1 2 2 2	72,25
3 3 2 2	104,

CONCENTRATION : 56.1 g Pu par l	
Configuration	Hc ∞ (cm)
1 1 1 1	127,
1 1 2 2	50,81
1 2 2 2	50,96
2 2 2 2	106,56
3 3 2 2	61,44

CONCENTRATION : 74.2 g Pu par l	
Configuration	Hc∞ (cm)
1 1 1 1	110,
1 1 2 1	51,22
1 1 2 2	43,1
1 2 2 2	43,1
2 2 2 1	76,37
2 2 2 2	57,1
3 3 2 2	48,85

Figure 6. Interaction entre cylindres annulaires.

Courbes $H_c/H_{c\,\infty}$ en fonction de la distance minimale entre solutions (H_c = hauteur critique des récipients en interaction dans une configuration donnée; $H_{c\,\infty}$ = hauteur critique d'un récipient isolé dans la même configuration)

Cylindre en interaction

Les courbes présentées font apparaître les effets d'interaction en vue directe, ou à travers de l'eau, et leur importance relative, pour diverses distances entre les solutions.

MÉTHODES DE CALCUL APPLICATIONS AUX EXPÉRIENCES

Les méthodes de calcul utilisées pour retrouver les résultats expérimentaux peuvent se diviser en trois catégories (méthode basée sur la théorie de la diffusion, à deux groupes ou à plusieurs groupes d'énergie, sur la théorie du transport – TDC 4 groupes –, et sur la théorie des probabilités – méthode de Monte-Carlo).

Méthodes basées sur la théorie de la diffusion Cylindres réfléchis et cylindres nus

Un autocode utilisant la méthode de la diffusion à deux groupes d'énergie (SEC 034) permet de retrouver les résultats de la cuve ∅ = 300 mm, diversement réfléchis. On donne dans [4] les sections efficaces utilisées

pour les divers réflecteurs envisagés de cette cuve, pour une solution de plutonium.

Les calculs effectués sur les cylindres d'ALECTO ($\varnothing = 420$ et $\varnothing = 300$) se prêtent à une interprétation plus directe, par ajustement d'une probabilité de fuites $P(B^2)$ sur les courbes expérimentales. La courbe de la figure 7 donne l'allure de cette fonction permettant de retrouver les résultats expérimentaux avec les sections efficaces précédentes.

Le code ALCI (SCEA 021 S) a été utilisé pour les cylindres d'ALECTO, avec un nombre de groupes d'énergie égal à 4. Les sections efficaces utilisées sont calculées à partir du code MUFT IV sur IBM 7090.

Méthodes basées sur la théorie du transport Cylindres pleins – Cylindres annulaires

Le programme utilisé est le programme TDC à deux dimensions utilisant la méthode Sn de Carlson. Le programme TDC a servi à calculer certains cas de géométrie d'ALECTO, ainsi que les cylindres annulaires testés à Dijon avec trou intérieur rempli d'air. Pour l'ensemble des cas, la comparaison entre valeurs expérimentales et calculées, traduites en coefficient de multiplication effectif, fait apparaître des différences inférieures à 1 %.

Méthode de Monte-Carlo [9]

Les problèmes pratiques rencontrés dans le domaine de la criticité présentent généralement une géométrie complexe, et ils ne peuvent être abordés par des théories de la diffusion ou du transport monodimensionnelles ou bidimensionnelles qu'en faisant des simplifications géométriques qui conduisent, en général, à surestimer excessivement la réactivité des systèmes. Les méthodes de Monte-Carlo utilisées pour la détermination de la réactivité des systèmes multiplicateurs ont l'avantage de pouvoir résoudre les systèmes complexes avec une grande fidélité géométrique. Elles sont donc largement employées dans le domaine de la criticité, et il était alors utile de voir la validité de ces méthodes appliquées aux cas expérimentaux étudiés.

Nous avons utilisé une méthode adaptée à la géométrie des cas étudiés. Les sections efficaces épithermiques sont issues de AWRE 0.28/60. Pour le domaine thermique, il y a un seul groupe et les sections efficaces pour les milieux multiplicateurs sont fournies par le code SOFOCATE, (hypothèses de Wigner et Wilkins). Pour les parois et le réflecteur (eau), des sections constantes ont été adoptées.

Etant donnés les temps de calcul par cas (20 à 30 min. sur ordinateur IBM 7094 suivant la concentration et pour une précision de 1 %) il n'était pas possible d'étudier toutes les expériences; on a donc fait un choix de façon à obtenir des résultats pour des géométries très différentes et des concentrations variées.

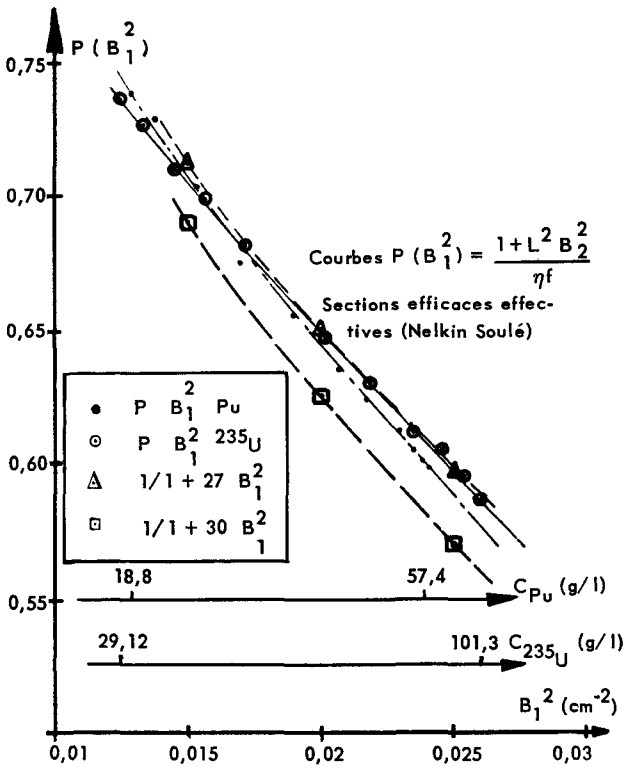


Figure 7. Ajustement réacteurs nus

Cuve ∅ 300 et 420 mm; solutions Pu et ^{235}U

Les cas calculés sont indiqués sur les courbes expérimentales (fig. 1, 2, 5, 6): ALECTO en cuve cylindrique de 300 et 420 mm de diamètre, nue et réfléchie; cylindres annulaires 500 × 300 mm; enfin, les cas en interaction dans l'eau et dans l'air (semi-réflexions). Pour les cas réfléchis par l'eau, une épaisseur minimale de 15 cm a toujours été prise.

Les résultats sont fournis directement en facteur de multiplication. Le rapport [9] donne les valeurs obtenues dans chaque cas avec les erreurs correspondantes. L'examen de l'ensemble des résultats ne fait pas apparaître de différences systématiques dues à la concentration: le fait que les résultats sur les cylindres pleins (cas nus) soient un peu inférieurs (1,5 %) est probablement dû à quelque réflexion par les parois latérales et supérieures de l'enceinte qui n'ont pas été considérées dans les calculs.

PRÉSENTATION DU GUIDE DE CRITICITÉ APPLICATION A QUELQUES CAS PRATIQUES

Présentation du guide de criticité

On a souvent rencontré des difficultés pour garantir la sûreté nucléaire d'une installation mettant en jeu des matériaux fissiles, lorsque les problèmes de criticité n'avaient pas été envisagés suffisamment tôt au moment de sa conception. Il est alors apparu souhaitable que les personnes chargées de l'élaboration de

telles installations, en général non spécialisées, disposent d'un instrument de travail qui leur permette, dès les premiers stades des études, de tenir compte des problèmes de sûreté nucléaire et de se référer à certaines normes; tel a été l'objectif principal de ce « guide de criticité » [10].

Le projet actuel comporte essentiellement des notions de base sur les réactions en chaîne, dans des milieux isolés ou en interaction, un exposé des notions fondamentales de sécurité nuclaire, mettant en lumière les principaux paramètres de contrôle utilisés. Enfin, un chapitre pratique complète ce guide. Il donne le processus à suivre pour examiner la sûreté d'une installation, en mettant en évidence un schéma général d'étude. Des courbes critiques et sous-critiques complètent ce travail.

Application à quelques cas pratiques

La résolution des problèmes de criticité pour l'appareillage industriel de l'usine de traitement des combustibles irradiés de La Hague a été faite en appliquant les principes indiqués ci-dessus.

Un des problèmes les plus importants concerne la sécurité des batteries d'extraction. L'usine comprend deux modèles de batteries de mélangeurs-décanteurs:

— Des appareils d'un volume moyen de 300 l avec 30 cm de hauteur totale pour les deux premiers cycles de purification au TBP de la solution uranium et plutonium.

— Des appareils d'un volume de 45 l avec 10 cm de hauteur totale pour le cycle de purification avec la trilaurylamine de la solution de plutonium après partition.

Pour les batteries de grand volume, le facteur principal de sécurité est nécessairement le contrôle de la concentration en plutonium; celle-ci est maintenue dans tous les étages à une valeur inférieure ou égale à 2 g/l, très éloignée de la valeur critique; dans ces conditions, les risques éventuels de criticité pour ces batteries très stables ne pourraient provenir que des risques d'accumulation lente de plutonium par dépôts solides. Les précautions suivantes ont donc été prises pour s'assurer que dans les opérations chimiques de telles conditions de précipitation ne peuvent se présenter:

— Vérification aussi complète que possible des schémas d'extractions sur un ensemble expérimental au 1/1 000 de l'usine (chaîne Gascogne de Fontenay).

— Adoption de schémas spéciaux pour éviter les accumulations de plutonium, en particulier dans l'extraction I, grâce à une section d'extraction complémentaire, et pour éviter l'hydrolyse du plutonium à la réextraction II par maintien d'une acidité minimum dans les appareils chargés.

— Contrôles α et neutroniques en plus du contrôle des débits et du contrôle analytique périodique.

Pour les batteries de purification finale du plutonium par une extraction à la trilaurylamine suivie d'une réextraction sulfurique, la marge de sécurité est aussi très importante; le contrôle de la criticité utilise ici le facteur géométrique et le facteur concentration de la solution en plutonium:

— La concentration critique de ces batteries a été vérifiée expérimentalement à la station de criticité de Dijon.

— Le schéma de purification a été vérifié à l'atelier pilote de traitement des combustibles irradiés de Marcoule à l'échelle 1/5.

— Tous les mélangeurs-décanteurs peuvent être munis de compteurs à neutrons type BF_3.

Par ailleurs, à l'usine de La Hague, toutes les batteries peuvent être vidangées intégralement et décontaminées et le contrôle du procédé est facilité par la centralisation des informations avec enregistrement numérique.

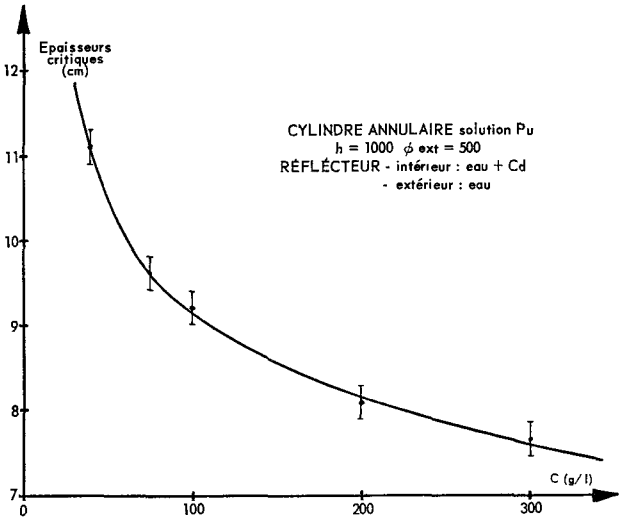


Figure 8. Epaisseur critique en fonction de la concentration

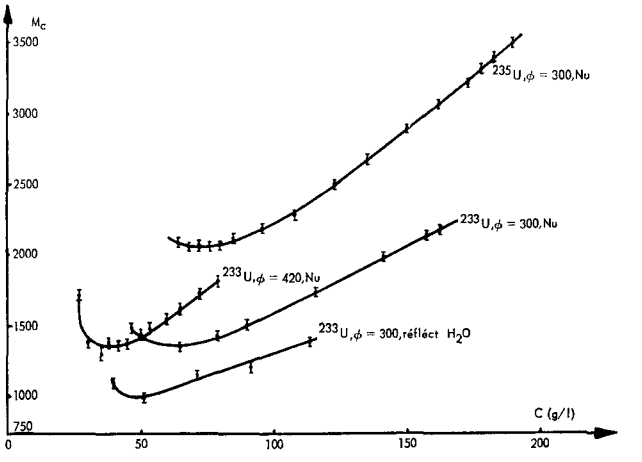


Figure 9. Masses critiques, ^{235}U et ^{233}U

Problèmes de stockage

Des expériences sur les réservoirs annulaires effectuées à Dijon, on a pu déduire un mode de calcul pour le stockage de solutions fissiles dans des configurations géométriques légèrement différentes et un domaine de concentration un peu plus étendu. Ainsi par exemple, la figure 8 donne l'épaisseur critique d'un réservoir annulaire ∅ ext = 500 réfléchi intérieurement avec feuille de cadmium sur la paroi interne [9].

CONCLUSION

Les études de criticité présentées ici constituent une partie des travaux entrepris par le CEA dans le domaine de la sûreté nucléaire. Il paraît important de souligner que la facilité d'exploitation des installations, leur rendement et leur prix de revient ne peuvent pas être dissociés de l'aspect sûreté. Les études de sûreté doivent donc être entreprises dès la conception des installations.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/84 France

Criticality studies

By D. Breton *et al.*

Criticality studies carried out by the French Atomic Energy Commission deal on the one hand with experiments on plutonium and enriched uranium solutions and on the other hand with theoretical work on the development and use of computation methods for the resolution of the nuclear safety problems of chemical and metallurgical plants.

Since 1958, the experimental studies have been focussed on homogeneous media composed of a fissile salt dissolved in light water. Developed on the PROSERPINE reactor, the experiments were continued at Saclay on the ALECTO assemblies, where solutions of plutonium or of 90%-enriched uranium can be made critical. The results already obtained relate to critical masses of cylindrical tanks of various diameters between 20 and 50 cm, reflected in several ways (by water, concrete, etc.) at concentrations of up to 100 g/l. Physical measurements (spectra, reactor noises) and interaction measurements supplement the chemical results.

Other experiments on plutonium solutions were begun in 1963 at the Valduc Centre. They deal with the study of critical masses of solutions kept in annular vessels of 50 cm external diameter and with an internal diameter of from 10 to 30 cm. These vessels can be reflected with water, either internally, or externally, or both internally and externally. The interaction of two such vessels has been studied for various geometries.

Slabs of various thicknesses were also studied.

The studies carried out have enabled methods of computation to be perfected which have been tried out in a number of practical ways. Particular use was made of the potentialities of calculations based on transport theory and on the Monte-Carlo method.

All these theoretical studies find application in the design and control of industrial plants from the safety angle.

A/84 Франция

Исследования критичности

Бретон *et al.*

Исследования критичности, проведенные в КАЭ, включают, с одной стороны, опыты с растворами плутония и обогащенного урана и, с другой стороны, теоретические работы, касающиеся разработки и применения расчетных методов для решения проблем ядерной безопасности, возникающих на химических и металлургических установках.

Опыты проводились начиная с 1958 года с растворами солей делящегося материала в обычной воде. Эти опыты, проведенные на реакторе PROSERPINE, были затем продолжены в Сакле на критических сборках ALECTO с растворами плутония и обогащенного до 90% урана. Полученные результаты относятся к критическим массам, набранным в цилиндрических баках диаметром от 20 до 50 *см*, имеющих различные отражатели (вода, бетон и т. д.) при концентрации, доходящей до 100 *г/л*. Физические измерения (спектров, шумов реактора) и измерения взаимодействия дополняют полученные результаты.

Другие опыты с растворами были начаты в Центре Вальдука в 1963 году. Исследовались критические массы, набранные в кольцеобразных резервуарах с внешним диаметром 50 *см* и с внутренним диаметром от 10 до 30 *см*. Эти критические сборки могут иметь отражатели внутри и снаружи или оба. Два таких резервуара были изучены во взаимодействии для различных конфигураций.

Также проводились опыты с плоскими баками различной толщины.

Проведенные исследования позволили разработать расчетные методы, которые были проверены на многочисленных экспериментальных данных. В частности, были проверены возможности расчета, основанного на теории переноса и на методе Монте-Карло. Все эти теоретические предпосылки были использованы при разработке и контроле промышленных установок с точки зрения их безопасности.

A/84 Francia

Estudios de criticidad

por D. Breton *et al.*

Los estudios de criticidad realizados en el CEA comprenden experimentos con soluciones de plutonio y de uranio enriquecido por una parte y, por otra, trabajos teóricos sobre la elaboración y la aplicación de métodos de cálculo destinados a resolver los problemas de seguridad nuclear planteados por las instalaciones químicas y metalúrgicas.

Los estudios experimentales versan desde 1958 sobre medios homogéneos compuestos por una sal fisible disuelta en agua ligera. Los experimentos se han preparado en el reactor PROSERPINE y han proseguido en Saclay en los conjuntos ALECTO, en los que pueden alcanzar la criticidad soluciones de plutonio o de uranio enriquecido al 90%. Los resultados obtenidos se refieren a las masas críticas de recipientes cilíndricos de diámetro comprendido entre 20 y 50 cm, con diferentes reflectores (agua, hormigón, etc.) en concentraciones de hasta 100 g/l. Los resultados obtenidos se complementan con mediciones físicas (espectros, actividad de fondos en el reactor) y de interacción.

En el Centro de Valduc se han emprendido en 1963 otros experimentos relativos a las soluciones de plutonio. Su objeto es estudiar la masa crítica de depósitos anulares de 50 cm de diámetro exterior, y de diámetro interior variable entre 10 y 30 cm. Para estos depósitos puede utilizarse agua, tanto en el interior como en el exterior, en calidad de reflector. Dos de ellos se han estudiado en interacción para diferentes configuraciones.

Otros experimentos han tenido como objeto el estudio de placas de diversos espesores.

Los estudios emprendidos han permitido elaborar métodos de cálculo que se han verificado en varios experimentos. Se han aprovechado particularmente las posibilidades de cálculo basadas en el método de transporte y en el de Monte-Carlo. Todos estos estudios teóricos se aplican al proyecto y control de las instalaciones industriales en el plano de la seguridad.

Détection des accidents de criticité et des ruptures de gaines au Commissariat à l'énergie atomique

par G. Debrié, J. Lavie, M. Planque, R. Le Meur, J. Mégy et A. Roguin *

DÉTECTION DES ACCIDENTS DE CRITICITÉ

(G. Debrié, J. Lavie, M. Planque)

Au cours de ces dernières années, la manipulation de quantités croissantes de matière fissile a amené le Commissariat à l'énergie atomique à se préoccuper des risques d'accidents de criticité et, en particulier, de leur détection. Des études de matériel adéquat ont été lancées et ont abouti à la réalisation d'un appareillage efficace. L'objet de cette communication est de présenter les critères de choix de la solution adoptée, les caractéristiques imposées au matériel, les réalisations et leurs essais ainsi qu'un exemple d'installation complète de détection d'accidents de criticité.

Nécessité d'une détection

Dans toute installation où est manipulée de la matière fissile en quantité potentiellement supérieure à la masse critique, subsiste quelles que soient les précautions prises, un risque permanent d'accident de criticité. Celui-ci peut consister soit en une seule excursion nucléaire, d'une durée inférieure à une fraction de seconde, soit en une réaction auto-entretenue d'une durée indéfinie, soit encore en une succession d'excursions par oscillations, pouvant entraîner une radioexposition externe grave du personnel. S'il est très difficile, sinon impossible, de prévenir de l'imminence d'un accident donc d'éviter l'irradiation due à l'excursion initiale, par contre, la détection immédiate de cette dernière, suivie d'une alarme et d'une évacuation rapide du personnel limite considérablement les risques d'irradiation et de contamination dus aux produits de fission ou à la réaction nucléaire elle-même si elle se poursuit. Les manifestations physiques d'une excursion critique: éclair bleu dû à l'effet Mallet-Cerenkov, bruit d'explosion, vapeurs nitreuses, odeur d'ozone, bien qu'elles doivent être connues des exploitants demeurent nettement insuffisantes pour donner l'alarme et peuvent par ailleurs prêter à confusion. La nécessité d'un appareillage approprié à la détection des accidents de criticité s'impose donc.

Critères de choix

Les installations retenues doivent présenter une grande sécurité de fonctionnement. Dans la solution choisie elles assureront uniquement la fonction de détection à l'exclusion des mesures, localisation, balisage de l'évacuation.

Un tel ensemble doit donc:

a) Détecter immédiatement toute excursion critique quelles que soient son amplitude et sa forme et cela uniquement dans sa zone de surveillance, afin de déclencher immédiatement l'alarme;

b) Présenter un haut degré de sécurité de fonctionnement avec une probabilité de panne la plus faible possible. En particulier, le taux de fausse alarme devra être limité par l'emploi de matériel électronique sûr.

Avant de traduire ces qualités essentielles sur le plan pratique, rappelons brièvement les caractéristiques d'un accident de criticité [1];

a) Une excursion nucléaire se manifeste entre autres, par l'émission de photons gamma et de neutrons— 10^{-14} rad/fission environ à un mètre pour chaque type de rayonnement;

b) L'amplitude du phénomène peut s'étendre de 10^{15} à quelque 10^{19} fissions, la valeur 10^{18} étant la plus probable;

c) Son évolution dans le temps est affectée d'un facteur de forme variable, dont la seule partie commune à tous les cas est l'existence d'une montée dont la durée minimale constatée est de 10^{-4} s et qui peut atteindre quelques secondes.

Il résulte de ces données et de l'objectif visé les conclusions suivantes:

a) La rapidité de réponse exige que l'alarme soit déclenchée à partir de l'émission gamma ou neutron lors de l'excursion nucléaire initiale;

b) La sensibilité des appareils doit leur permettre de détecter l'excursion critique minimale sans toutefois être éblouis lors de l'accident maximal;

c) Afin de limiter le rayon de la zone de déclenchement de l'alarme et de s'affranchir de l'incertitude qui entoure la durée d'une excursion critique, la sonde de

* Commissariat à l'énergie atomique.

détection doit déclencher pour une dose intégrée définie et non pour une intensité prédéterminée;

d) La sécurité de fonctionnement nécessite l'adoption des principes ci-dessous:

i) Toute panne concevable des sondes de détection doit, dans la mesure du possible donner lieu à un signal de défaut identique à celui d'alarme;

ii) L'alarme ne doit être déclenchée que par coïncidence des informations en provenance d'au moins deux sondes de détection surveillant le même appareillage siège de la divergence, ce dernier étant par conséquence surveillé par au moins trois détecteurs afin d'assurer une continuité dans le contrôle en cas de panne de l'un d'entre eux.

Caractéristiques adoptées

A partir de ces principes de base et à la demande de la Sous-Commission des masses critiques du Commissariat à l'énergie atomique, des caractéristiques physiques et électroniques minimales d'un matériel garantissant un fonctionnement sûr et réduisant les risques de fausses alarmes ont été établies. Les spécifications essentielles retenues sont les suivantes [2]:

Sondes de détection

Elles doivent détecter soit le rayonnement gamma soit les neutrons émis lors de l'excursion nucléaire initiale;

Leur seuil doit permettre le déclenchement de l'alarme lorsque les détecteurs auront reçu une dose intégrée correspondant à 10^{14} fissions à la source avec un rayon d'action suffisant (30 m avec une atténuation de 10 par les différents obstacles);

Elles seront insensibles à un niveau permanent d'irradiation neutron ou gamma inférieur ou égal à 100 m rad/h;

Elles ne doivent pas être inhibées par une intensité d'irradiation gamma correspondant à 10^5 rad en 1 milliseconde;

Elles doivent pouvoir subir une irradiation correspondant à 10^{20} fissions à 1 m de la source sans être détériorées;

Le délai total entre le franchissement du seuil et l'apparition du signal de sortie doit être inférieur à 1 milliseconde;

Toute panne concevable doit donner lieu, dans la mesure du possible, à un signal de défaut analogue à celui de dépassement de seuil;

Elles doivent engendrer elles-mêmes le signal de sortie constitué par une tension continue qui s'annulera en cas de dépassement de seuil;

Elles doivent être pourvues d'un dispositif de contrôle global de l'ensemble détecteur électronique associé;

Elles seront groupées en unités de surveillance dont le nombre et l'implantation seront fixés compte tenu de la distribution des locaux et des risques. Chaque unité de coïncidence comprendra un minimum de 3 sondes.

Ensemble de traitement d'information

L'alarme ne sera donnée qu'en cas de coïncidence, dans un intervalle de temps inférieur à 500 ms, des signaux d'alerte en provenance d'au moins deux sondes d'une même unité de surveillance;

L'alarme sonore sera déclenchée au plus tard 500 ms après l'apparition d'une information d'alarme;

Toute sonde ayant donné un signal d'alerte sera automatiquement isolée de sa chaîne de coïncidence si dans un intervalle de 500 ms, une deuxième sonde n'a pas donné d'information. Le défaut sera alors affiché au tableau de contrôle.

Caractéristiques communes

Une autonomie de 12 h en cas de coupure secteur sera assurée pour l'ensemble de l'installation par batteries « tampon »;

L'installation devra répondre aux spécifications du matériel de sécurité en ce qui concerne la sensibilité aux variations secteur, aux parasites secteur, à la température.

Matériel étudié au CEA

Trois principes différents de détection ont été envisagés: la chambre d'ionisation, l'ensemble scintillateur-photomultiplicateur, le détecteur semi-conducteur. Chacun de ces principes a fait l'objet d'une ou de plusieurs réalisations qu'il a été possible — grâce à l'obligeance de la Health Physics Division à ORNL — de tester en présence de « flashes » réels, d'amplitudes et de formes identiques à celles qui peuvent être rencontrées en cas d'accidents. Les résultats très satisfaisants des essais ont permis de retenir les deux derniers principes de détection mentionnés ci-dessus, les détecteurs à chambre d'ionisation ne s'étant pas avérés suffisamment sensibles.

Sonde de détection à photomultiplicateur

Essentiellement sensible au rayonnement gamma, elle est composée d'un scintillateur TPB associé à un photomultiplicateur et de circuits électroniques à transistors. Le schéma synoptique de cette sonde est montré (fig. 1). Deux particularités sont à signaler dans cet appareil:

a) Un relais excité à partir du tableau de contrôle de l'installation permet de tester la sonde en envoyant une brève impulsion de tension sur une lampe à filament placée en vue du photomultiplicateur;

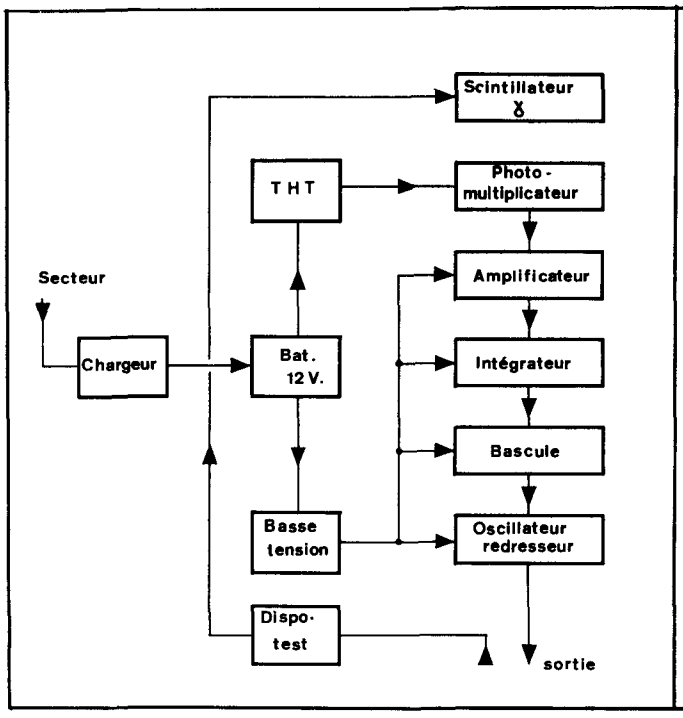


Figure 1. Sonde à photomultiplicateur

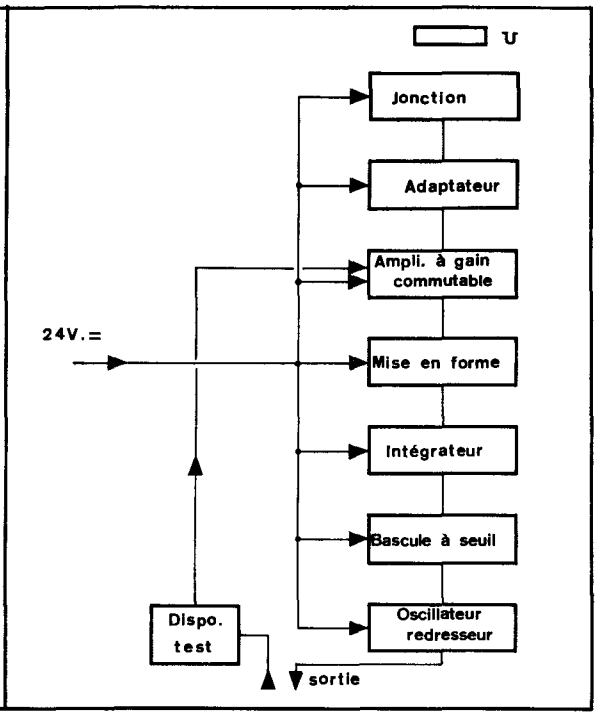


Figure 2. Sonde à détecteur semi-conducteur

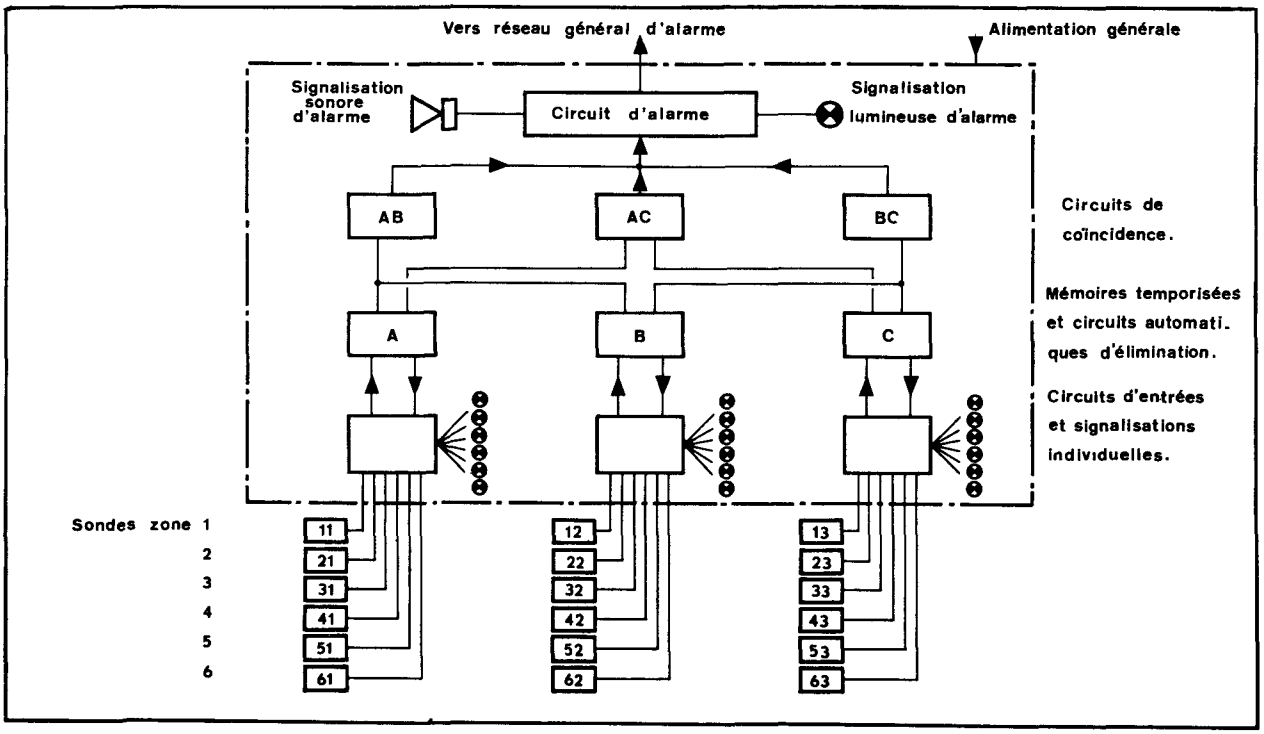


Figure 3. Ensemble de traitement des informations

b) Un circuit de sécurité permet de signaler tout fonctionnement anormal de la très haute tension.

Les essais à ORNL ont révélé une très grande sensibilité de ce type de sonde, meilleure que 0,08 mrad.

Sonde à détecteur semi-conducteur

L'élément sensible est une jonction NP au silicium en regard, et à faible distance de laquelle est disposée une pastille d'uranium enrichi. La détection s'effectue par comptage des noyaux de recul de fission de l'uranium, produits sous l'action des neutrons émis pendant l'accident. En plus de cet élément détecteur, la sonde comprend un certain nombre de circuits électroniques nécessaires à son fonctionnement (fig. 2).

Le test s'effectue en commutant à distance le gain d'un étage amplificateur, permettant alors la détection des particules alpha émises spontanément par l'uranium, les impulsions dues à ces dernières étant d'amplitude plus faible que celles dues aux noyaux de recul. La possibilité d'alimenter les circuits à transistors ainsi que la jonction sous une tension de 24 V permet d'envisager l'alimentation de l'ensemble des sondes à partir d'une batterie d'accumulateurs unique.

Lors des essais aux Etats-Unis, la sensibilité de cette sonde a été trouvée meilleure que 0,2 mrad.

Ensemble de traitement des informations

Afin de réduire le nombre de fausses alarmes, le principe de la coïncidence de 2 informations parmi n a été retenu. Compte tenu du taux de pannes constaté pour les sondes et d'un contrôle journalier de l'installation, la probabilité de fausse alarme demeure extrêmement faible à partir de $n = 3$. En revanche, la multiplication du nombre d'éléments constitutifs qui entraîne obligatoirement un accroissement du taux de pannes et du coût de l'installation milite en faveur d'un nombre n faible. L'élimination automatique d'une sonde en défaut conduit à adopter $n \geqslant 3$ pour que l'installation continue à fonctionner. Cependant, la majeure partie des installations nécessite, compte tenu de la disposition géométrique des lieux, un nombre relativement important de détecteurs. L'étude et la standardisation d'un ensemble de traitement des informations basé sur le principe de la coïncidence de 2 parmi 3, mais pouvant recevoir sur chacune de ses 3 entrées les informations de 6 sondes, a permis de réduire notablement l'importance du matériel nécessaire dans chaque installation où le nombre de zones à surveiller est important. Le schéma synoptique d'un tel appareillage est montré (fig. 3). L'ensemble est complété par un dispositif de test qui permet de vérifier le bon fonctionnement des sondes, des circuits de coïncidence, de l'élimination automatique des détecteurs en défaut. Durant les opérations de test, la priorité est laissée à une véritable alerte.

Chaque installation est alimentée à partir de batteries cadmium-nickel en tampon assurant une autonomie de 12 h en cas de coupure secteur. Par mesure de sécurité, les organes vitaux sont doublés avec commutation automatique en cas de défaillance décelable.

Exemple d'installation

La figure 4 présente un exemple complet de mise en pratique des principes généraux énumérés ci-dessus.

Le bâtiment a pu être divisé en deux parties principales. La zone A, constituée de locaux séparés par des cloisons minces, a nécessité la mise en place de 3 sondes de détection pour une superficie de 1 200 m^2. Par contre, la zone B, dans laquelle sont disposées 12 cellules, a nécessité 12 sondes pour une superficie de 1 800 m^2. Les 15 sondes sont reliées à un châssis unique de traitement des informations. Le mode de groupement de ces sondes est indiqué sur la figure.

Conclusion

La difficulté du problème posé par la détection des accidents de criticité imposait une coordination des études entres les différents services intéressés du CEA. Cette coordination a abouti à la définition et à la réalisation d'un matériel standardisé répondant à l'objectif fixé. Il convient cependant de souligner que quelles que soient ses qualités, l'appareillage de détection reste impuissant à déceler l'imminence d'une réaction en chaîne et que tout doit donc être fait pour rendre l'installation aussi sûre que possible.

DÉTECTION DES RUPTURES DE GAINES
(R. Le Meur, J. Megy et A. Roguin)

Perspectives générales

Les centrales nucléaires n'échappent pas aux lois de la thermo-dynamique et l'augmentation du rendement a été obtenu par une augmentation de la température et de la pression du fluide gazeux de refroidissement; le rendement escompté pour EDF3 est de 31 % [3]. D'autre part, l'utilisation de l'oxyde d'uranium comme combustible dans les piles refroidies par gaz permet d'atteindre de très hautes valeurs du taux d'irradiation: 10 000 MW j/t pour EL4 [4].

Ces progrès dans l'utilisation de l'énergie nucléaire ont pu être réalisés grâce aux efforts immenses dirigés vers les études thermiques et métallurgiques des éléments combustibles. Malgré les résultats très encourageants obtenus, il n'est pas possible de garantir que l'élément combustible pourra toujours contenir la très forte radioactivité de ses produits de fission. D'autre part, il est évident que, pour un combustible donné, la marche la plus économique d'une centrale est celle qui admet en moyenne un certain nombre de ruptures de gaines par mois. Les combustibles détériorés doivent

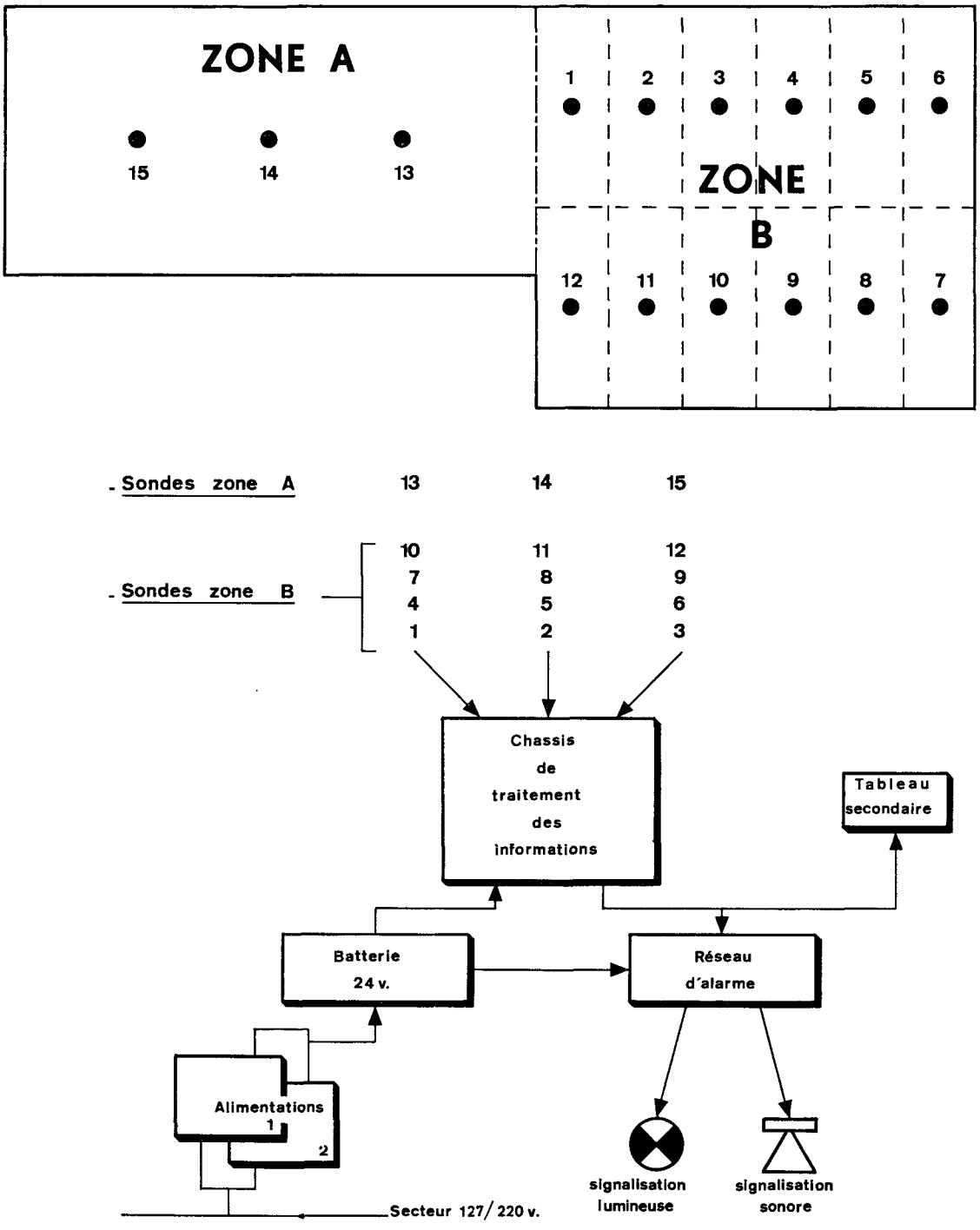


Figure 4. Schéma complet d'une installation

malgré tout être déchargés avant la pollution radioactive des circuits de refroidissement. Les installations de détection des ruptures de gaines (DRG), dont le rôle et l'utilité se sont confirmé ces dernières années, doivent donc suivre le progrès des centrales et s'adapter constamment aux nouvelles conditions de fonctionnement. Cette évolution s'est traduite essentiellement par l'étude de détecteurs de produits de fission mieux adaptés et plus sensibles et par la recherche de l'automatisme des installations assurant une exploitation facile et sûre.

Dans cet article, on traitera d'abord de l'évolution des détecteurs et des équipements du traitement des informations dans les installations de DRG. Quelques nouvelles installations seront ensuite succintement décrites.

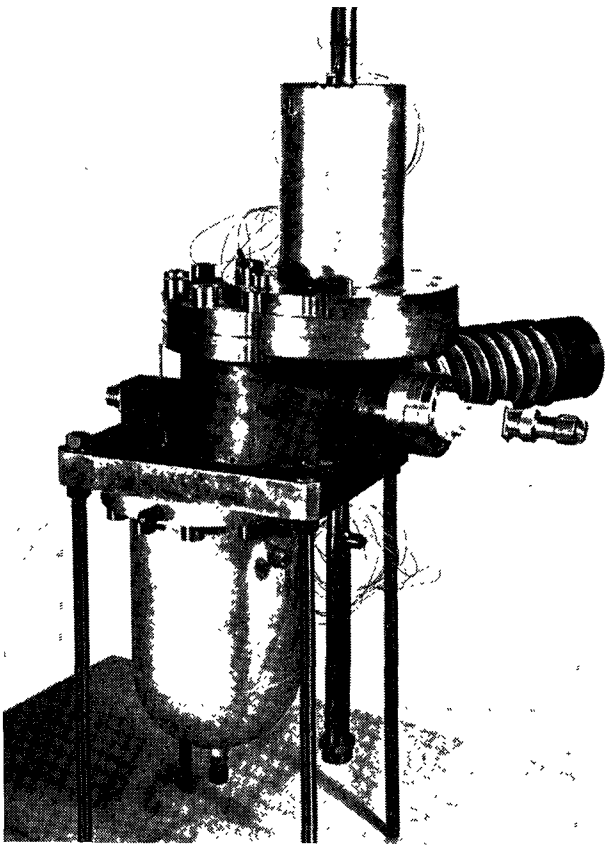


Figure 5

Les détecteurs de produits de fission

a) Le détecteur des produits de fission utilisé pour la DRG de la pile EL4 est basé sur le même principe que ceux utilisés pour les piles G3 et la filière EDF de Chinon [5]: ils mesurent la radioactivité ß des atomes rubidium et césium descendants des gaz de fission, après leur extraction du fluide caloporteur.

L'utilisation de gaz carbonique à la pression de 60 bars a toutefois posé de nouveaux problèmes physiques et technologiques. Pour les résoudre une boucle avec circuit en pile a été construite transportant des gaz de fission dans du gaz carbonique à 60 bars et à 400 °C. Cette boucle a permis d'essayer plusieurs types de détecteurs de produits de fission dont un prototype adapté à l'installation de DRG prévue pour EL4 (fig. 5):

i) Aux fortes pressions la mobilité des ions est très faible, sa mesure lors des essais du prototype a donné: 8.10^{-3} cm/s par V/cm dans le gaz carbonique à la pression de 50 bars et à 20 °C. Le champ électrique nécessaire à la précipitation doit pouvoir atteindre 30 kV par cm dans la chambre de collection soit une haute tension maximale de 15 000 V.

ii) A l'inverse des autres installations où l'on refroidissait le gaz prélevé, ici il doit être maintenu chaud, car à 60 bars, CO_2 se liquéfie à la température de 21 °C. La température du gaz traversant le détecteur sera donc d'environ 200 °C et tous les matériaux du détecteur ont été choisis pour avoir un comportement sûr à cette température et à cette pression: en particulier l'isolant de l'électrode HT constitué par un tube d'alumine frittée brasé sur du titane.

iii) Par suite de la température de fonctionnement il a fallu utiliser un matériau scintillant qui conserve ses propriétés de scintillation constantes jusqu'à 200 °C au moins. Un verre scintillant a donné de très bons résultats; il est associé à un conduit de lumière en quartz qui supporte la pression et transmet les impulsions lumineuses à un photomultiplicateur.

Le détecteur prototype essayé dans ces conditions a une très bonne sensibilité. Installé à EL4 il permettra d'afficher une alarme pour des ruptures de gaine de surfaces équivalentes émissives inférieures à 4 mm² de combustible nu.

b) Le comportement des ruptures de gaine des crayons chargés en oxyde d'uranium est encore inconnu et l'on peut craindre que dans certains cas des détecteurs à collection électrique ne puissent identifier aisément le canal défectueux si une forte bouffée de gaz de fission s'échappe par la rupture. Ainsi, dans un but expérimental, un détecteur de neutrons a été placé à proximité d'un filtre en bronze fritté, sélectif pour les iodes et les bromes, et situé en amont du détecteur à collection électrique (fig. 6).

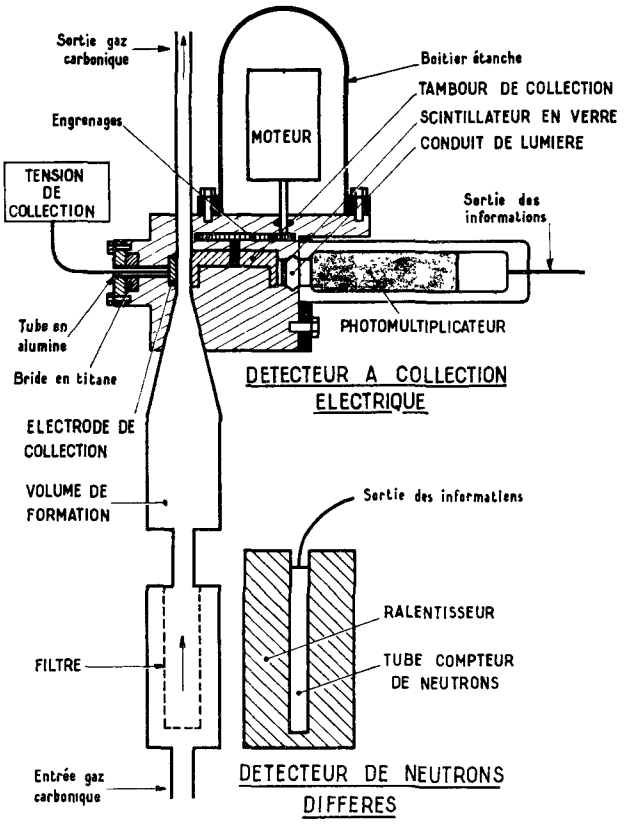


Figure 6. Association détecteur à collection électrique et détecteur de neutrons différés

De cette façon les produits de fission précurseurs de neutrons différés et particulièrement ^{137}I et ^{87}Br s'échappant lors d'une rupture de gaine seront mesurés et confirmeront l'identification du canal possédant un élément défectueux.

Des essais effectués sur la boucle ont montré que dans le gaz carbonique sous pression et à 200 °C cette méthode donnait un seuil de détection supérieur à celui obtenu par collection électrique, mais néanmoins très acceptable d'environ 8 mm² de combustible nu.

c) Un nouveau type de détecteur a été mis au point pour la mesure de la concentration des gaz de fission dans les circuits de gaz à basse pression (1 à 6 bars) (fig. 7).

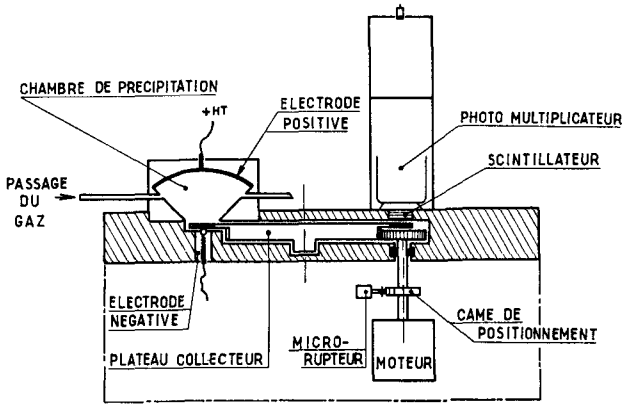


Figure 7. Détecteur à collection instantanée en géométrie sphérique: type à plateau

L'électrode de collection est constituée par la surface plane d'un plateau animé des mêmes mouvements que le tambour des détecteurs déjà connus [6]. La chambre de formation des ions est constituée par un secteur sphérique parcouru par le gaz à analyser. Le champ électrique est établi dans cette chambre entre une électrode en forme de calotte sphérique et la surface du plateau.

La sensibilité de ces détecteurs n'est pas influencée par la vie moyenne des ions, elle est sensiblement indépendante du débit de gaz traversant le détecteur.

Automatisme et traitement des mesures

Le schéma d'exploitation de la DRG à EL4, comme celui de la pile G3 [7] comprend deux étages, l'un de « prospection » de tous les canaux groupés par faisceaux de 4, l'autre d'« identification » et d'« observation continue » des défauts. Des électrovannes 3 voies et 2 voies commutent des prélèvements de gaz à 60 bars et 400 °C. De nombreux essais ont permis de mettre au point une classe de matériel pneumatique de haute sécurité de fonctionnement.

Depuis la construction de la pile G3 qui possède une installation de DRG très automatique et toujours moderne par sa facilité d'exploitation [8], l'évolution des installations a été marquée surtout par les progrès dans le traitement des mesures et l'automatisme associé.

Mentionnons toutefois le mode d'exploitation de la DRG prévu sur la pile EDF3 et qui utilise un groupement matriciel des prélèvements gazeux des 3 260 canaux. La localisation de la rupture de gaine est faite par deux détecteurs explorant respectivement les lignes et les colonnes d'un groupe de 3 matrices carrées de 64 canaux.

Le traitement des mesures de la DRG de la pile EL4, de même que celui de la pile EDF3, sera effectué par un calculateur central qui traitera toutes les mesures de la pile et élaborera toutes les alarmes et signalisations [9].

Les mesures de prospection qui étaient autrefois systématiquement enregistrées seront maintenant ignorées tant que l'évolution de ces mesures ne dépassera pas un seuil de sécurité fixé. Ce seuil pour la plupart des combustibles peut être fixé à quelques millimètres carrés de surface équivalente émissive.

Toutefois, une carte complète des activités des produits de fission de tous les canaux peut être enregistrée à la demande.

Dès que le seuil de détection est dépassé en « prospection » et que le faisceau est pris en charge par le détecteur « suiveur », les mesures sont imprimées automatiquement sur machine à écrire ou inscrites sur un enregistreur. Ainsi, le conducteur de pile observe l'évolution de la fissure de gaine et peut prendre toute décision nécessaire: de déchargement immédiat, de déchargement sur programme normal, ou exceptionnellement d'une baisse de puissance de la pile.

Installation de DRG de quelques piles françaises

Le détecteur de produits de fission des boucles d'essais de combustible du réacteur PÉGASE est l'appareil essentiel pour le contrôle et la surveillance de la gaine donc du comportement des barreaux de combustibles en cours d'essais.

Les détecteurs ont été étudiés spécialement pour avoir une grande sensibilité et mesurer l'émission de produits de fission bien avant que la surface émissive équivalente n'atteigne 1 mm² [10, 11].

Bien que les ruptures de gaines soient moins fréquentes et aient des conséquences moins importantes dans les piles piscines, elles ne sont pourtant pas inexistantes. Le cœur de la pile PÉGASE a été équipé d'une installation de mesure des gaz de fission par collection électrique, après extraction de ceux-ci de l'eau de refroidissement. La pile SILOÉ possède une installation similaire à celle du cœur de PÉGASE.

Pour les autres piles refroidies par eau, la méthode de mesure des neutrons différés a été utilisée. Dans la pile OSIRIS en construction, un prélèvement d'eau effectué immédiatement à la sortie du cœur passe au centre d'un bloc de polyéthylène dans lequel sont

noyés 4 compteurs à BF_3. Deux chaînes électroniques — l'un à réponse linéaire, l'autre à réponse logarithmique — pourront suivre tous les cas possibles de défaut d'étanchéité des éléments.

Depuis sa mise en service, la pollution interne de la pile EL3 a augmenté, diminuant ainsi notablement la sensibilité de l'installation de DRG. Pour rétablir la sécurité de fonctionnement, une nouvelle installation est en cours de réalisation. Elle est basée également sur la mesure des neutrons différés. Les 100 prélèvements d'eau venant des cellules sont séparés en 10 groupes. Les 10 tubes de prélèvement d'un groupe sont disposés selon une cage d'écureuil dont le centre est occupé par un compteur à BF_3; l'ensemble est entouré par un bloc de modérateur en polyéthylène. Cette disposition permet une détection très rapide d'une rupture de gaine. La cellule défectueuse est ensuite identifiée en dérivant successivement chacun des 10 prélèvements du groupe vers un détecteur « suiveur » au moyen d'électrovalves 3 voies placés en amont des blocs « détecteurs de groupe ».

La pile RAPSODIE sera également équipée d'une installation de DRG. Deux chaînes sont prévues: l'une détectera les neutrons différés émis par les produits de fission véhiculés par le sodium, l'autre les gaz de fission mélangés à l'argon de couverture de la cuve.

Conclusion

Les méthodes de détection des produits de fission dans le fluide caloporteur des réacteurs sont maintenant très connues. La mise au point de détecteurs utilisant ces méthodes et le développement industriel d'appareillages pneumatiques ont permis de réaliser les installations indispensables de détection des ruptures de gaines. Les études en cours ont pour but d'adapter les procédés connus aux problèmes posés par les nouveaux réacteurs en projet.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/85 France

Detection of criticality accidents and burst cans as practised by the French Atomic Energy Commission

By G. Debrié *et al.*

Detection of criticality accidents

In all plants where fissile materials are handled in quantities potentially greater than the critical mass, there is a permanent risk of an accidental nuclear excursion entailing a serious irradiation hazard to personnel, no matter what precautions may have been taken. Immediate detection followed by rapid evacuation greatly limits the risk of irradiation from the fission products or from the nuclear reaction itself, if this is sustained. The need for suitable equipment for detecting criticality accidents arises from the fact that the physical phenomena accompanying a nuclear excursion are insufficient in themselves to give the alarm, and may lead to confusion.

Taking into account features of accidents that have occurred in the past, and the function expected of a detector, a dose-integration device was chosen for the detecting probe proper, and a system based on the coincidence of a number of data to trigger the alarm. At the request of the Commission's Sub-Committee on Critical Masses, the minimal physical and electronic features required to provide a reliable system with minimal risk of false alarm have been established. These features are described in some detail.

The equipment built at the French Atomic Energy Commission on the basis of these recommendations

has been tried out systematically under actual conditions of nuclear excursion at the the Health Physics Research Reactor of the National Laboratory at Oak Ridge. The tests made it possible to select two types of detection probe (which are described): a photomultiplier, and a semi-conductor probe.

Finally, the authors present a practical example of a complete unit for the detection of criticality accidents in an installation for processing fissile material.

Detection of burst cans

Most of the work done on burst slug detection (BSD) in recent years has been focussed on large gas-cooled reactors where the problem of burst slugs has a bearing on plant economics. The methods developed in this programme can be adapted to the easy solution of problems of detection in other types of reactor.

The general trend of evolution of the various installations already built, or under construction, has been towards a high degree of automation and the use of detectors capable of operating at higher and higher temperatures and under greater and greater pressures.

The successors to the G3 reactors—whose BSD installation was already highly automatic—the EDF reactors at Chinon, use digital computers especially designed for BSD-data processing. In the EL4 reactor, now at the design stage, the BSD measurements will be processed, together with all other reactor control measurements, in a centralized computer unit.

The steady increase in cooling-gas pressure (15 bars in G3, 25 bars in the Chinon reactors and 60 bars in EL4) has called for the development of electric-precipitation detectors capable of separating and measuring the fission products in the carbon dioxide at temperatures up to 300 °C and under pressures as high as 60 bars.

Despite the severe operating conditions, the various devices developed offer every guarantee of the reliability and long life required for use in BSD installations.

A/85 Франция

Обнаружение нарушений критичности и герметичности оболочек

Г. Дебре *et al.*

Обнаружение нарушений критичности

Несмотря на все меры предосторожности, в любой установке, на которой работают с делящимся материалом в количестве, потенциально превышающем критическую массу, существует постоянная опасность превышения номинального значения реактивности, которое может повлечь за собой облучение персонала. Немедленное обнаружение этого превышения и последующее быстрое удаление людей значительно снижают опасное действие излучения от продуктов деления или от самой ядерной реакции. Поскольку физические явления, сопровождающие отклонения реактивности, бывают недостаточными, чтобы вызвать тревогу, и могут дать повод к сомнениям, то необходима соответствующая аппаратура для обнаружения нарушений критичности.

Учитывая характеристики возможных нарушений и предполагаемую роль детектора, в качестве последнего был выбран интегрирующий дозиметр, а система для включения тревожной сигнализации была создана на принципе совпадений нескольких сигналов.

Несколько подробнее изложены минимальные физические и электронные характеристики устройства, гарантирующие надежную работу и максимально ограничивающие вероятность ложных тревог, в соответствии с требованием Комиссии по критическим массам КАЭ Франции.

Эти устройства, разработанные Комиссариатом по атомной энергии, были тщательно испытаны в реальных условиях на реакторе для изучения радиационной безопасности в Окриджской национальной лаборатории. На основании этих опытов были приняты два типа детекторов: детектор с электронным фотоумножителем и полупроводниковый детектор.

В заключение авторы описывают конкретный пример создания детектора нарушений критичности в установке, на которой перерабатывают делящийся материал.

Обнаружение нарушений герметичности оболочки

Основные работы в области контроля герметичности оболочки (КГО) проводились в последние годы применительно к крупным реакторам с газовым охлаждением, где проблема нарушения герметичности оболочки связана с рентабельностью атомной электростанции. С помощью метода, разработанного для этих реакторов, можно легко решить проблемы контроля герметичности в реакторах других типов.

Совершенствование различных систем КГО, созданных и создаваемых в настоящее время, характеризуется постоянным стремлением к более полной автоматизации и использованием детекторов, работающих при более высоких температурах и давлениях.

Если реактор G-3 уже имеет высокоавтоматизированную систему КГО, то для реакторов EDF в Шиноне предусматривается применение вычислительных машин, специально предназначенных для систем КГО. Для реакторов EL-4 разработан проект, по которому информация системы КГО вместе с информацией контрольно-измерительных приборов реактора будет обра-

батываться в центральной вычислительной установке.

Последовательное увеличение давления газа теплоносителя (15 *бар* в реакторе G-3, 25 *бар* в реакторах в Шиноне, 60 *бар* в реакторе EL-4) потребовало разработки детекторов с электростатическим фильтром, который может отделять и измерять продукты деления, содержащиеся в углекислом газе, при температурах до 300° С и при давлении до 60 *бар*.

Несмотря на жесткие условия эксплуатации, были созданы различные устройства, которые гарантируют безопасность и долговечность работы систем КГО.

A/85 . Francia

Detección de accidentes de criticidad y de rotura de vainas en el Commissariat à l'énergie atomique

por G. Debrié *et al.*

Detección de accidentes de criticidad

En toda instalación en la que se maneja material fisible en cantidad potencialmente superior a la masa crítica, existe, cualesquiera que sean las precauciones tomadas, un riesgo permanente de excursión nuclear accidental que puede llevar consigo una irradiación grave del personal. La detección inmediata seguida de una evacuación rápida limita considerablemente los riesgos de irradiación debidos a los productos de fisión o a la misma reacción nuclear si ésta continúa. Puesto que los fenómenos físicos que acompañan una excursión nuclear pueden ser insuficientes por sí mismos para producir alarma directa y pueden dar lugar a confusión, se hace necesario disponer de una instrumentación adecuada para la detección de accidentes de criticidad.

Teniendo en cuenta las características de los accidentes acaecidos y lo que se puede esperar de un detector, la elección recae sobre un dispositivo de integración de dosis en lo que concierne a la sonda de detección propiamente dicha y sobre un principio de coincidencia de varias informaciones para el disparo de la alarma. Se exponen con algún detalle las características físicas y electrónicas mínimas establecidas a petición de la Subcomisión de masas críticas del CEA, de un dispositivo que garantice un funcionamiento seguro y que limite al máximo el riesgo de falsas alarmas.

Los dispositivos realizados con estos requerimientos en el Commissariat à l'energie atomique han sido objeto de pruebas sistemáticas en condiciones reales de excursión nuclear en el Health Physics Research Reactor del Laboratorio Nacional de Oak Ridge. Estas pruebas han permitido retener como apropiados dos principios de sondas de detección que serán descritos: sonda de fotomultiplicador funcionando en corriente y sonda de semiconductor.

Finalmente los autores presentan un ejemplo concreto de realización de un equipo completo de detector de accidentes de criticidad en una instalación que emplea material fisible.

Detección de rotura de vainas (D. R.V.)

La mayoría de los esfuerzos en el campo de la detección de rotura de vainas han sido llevados a cabo estos últimos años en relación con los grandes reactores refrigerados por gas, en donde el problema de rotura de vainas está ligado a la rentabilidad de la central. Una adaptación de las técnicas puestas a punto en esta ocasión permite resolver fácilmente los problemas de detección para otros tipos de reactores.

Lo esencial de la evolución de las diversas instalaciones realizadas o en curso de realización se traduce en un progreso constante hacia una mayor automatización y en la utilización de detectores que funcionan bajo temperaturas y presiones cada vez más elevadas.

A continuación del reactor G3 que ya poseía una instalación muy automatizada, se ha visto utilizar para los reactores de la EDF (Electricité de France) de Chinon calculadoras digitales destinadas especialmente a la D.R.V. Para el reactor EL4 en proyecto, las medidas de la D.R.V. se tratarán juntamente con todas las medidas relativas al control del reactor en una unidad de cálculo centralizado.

El aumento progresivo de la presión del gas portador de calor (15 bares en el G3, 25 bares en los reactores de la central de Chinon, 60 bares en el EL4) ha hecho necesario la puesta a punto de detectores de colección eléctrica que puedan separar y medir los productos de fisión presentes en el gas carbónico, a temperaturas que llegan hasta 300 °C y bajo presiones que alcanzan los 60 bares.

A pesar de las severas condiciones de utilización, los diversos aparatos puestos a punto ofrecen todas las garantías de seguridad y de larga vida requeridas para su utilización en las instalaciones de D.R.V.

Критические параметры водных растворов $UO_2(NO_3)_2$ и ядерная безопасность

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1. КРИТИЧЕСКИЕ ПАРАМЕТРЫ ВОДНЫХ РАСТВОРОВ $UO_2(NO_3)_2$

Эксперименты по определению критических параметров водных растворов $UO_2(NO_3)_2$ проводились на сборках, имеющих форму цилиндров, прямоугольных параллелепипедов с квадратным основанием и сфер с водяным отражателем и без отражателя. Корпуса изучаемых сборок были выполнены из нержавеющей стали типа 1Х18Н9Т и имели толщину $1 \div 1{,}5$ *мм* для сфер и цилиндров и 3 *мм* для параллелепипедов.

При проведении экспериментов без отражателей минимальное расстояние активной зоны от бетонных стен помещения составляло 2 *м*.

Температура растворов изменялась в пределах $15 \div 21°$ С, а при проведении каждого отдельного эксперимента она изменялась приблизительно на $\pm 0{,}5°$ С.

Растворы $UO_2(NO_3)_2$ в воде содержат азотную кислоту. Величина отношения ядер $\beta_N = \rho_N/\rho_U$ в растворах изменялась во время проведения экспериментов при всех обогащениях и концентрациях урана в пределах $2{,}4 \div 3{,}2$. Для растворов урана 90%-ного обогащения можно считать, что изменения критических параметров растворов $UO_2(NO_3)_2$ за счет изменения температуры растворов за время проведения экспериментов не превышают ошибок, обусловленных точностью определения объемов раствора и концентраций урана, которая составляет $\pm 0{,}5\%$.

Экспериментальный материал по критическим массам сфер и параллелепипедов позволяет сделать заключение, что длина экстраполяции и экономия водяного отражателя зависят от концентрации урана и формы реактора.

Результаты опытов с параллелепипедами приводятся в табл. 1. Для каждой из трех концентраций урана в растворе было выполнено пять экспериментов на параллелепипедах. Среднеквадратичные значения геометрических параметров и длин экстраполяции для параллелепипедов были получены из экспериментов обработкой по методу наименьших квадратов. Ошибка в определении геометрического параметра не превышает $\pm 2\%$.

Таблица 1. Результаты критических экспериментов и расчетные значения геометрических параметров $\overline{B^2}$ и длин экстраполяции $\overline{d}$ для прямоугольных параллелепипедов с квадратным основанием без отражателя, содержащих растворы $UO_2(NO_3)_2$ 90%-ного обогащения

Размер стороны основания, *см*	Концентрация U^{235}, *г/л*	Отношение ρ_H/ρ_5	Отношение ρ_N/ρ_U	Критическая высота, *см*	$\overline{B^2}$	$\overline{d}_{пар}$
26,66	73,2	375,1	3,12	46,18		
28,45	73,2	375,1	3,12	36,71		
30,08	73,2	375,1	3,12	31,730	0,0222	2,71
32,63	73,2	375,1	3,12	27,40		
35,55	73,2	375,1	3,12	24,5		
26,66	152,3	175,7	3,08	29,01		
28,45	152,3	175,7	3,08	25,67		
30,08	152,3	175,7	3,08	23,59	0,02679	2,63
32,63	152,3	175,7	3,08	21,39		
35,55	152,3	175,7	3,08	19,76		
26,66	233,45	109,4	3,13	27,079		
28,45	233,45	109,4	3,13	24,20		
30,08	233,45	109,4	3,13	22,32	0,0279	2,67
32,63	233,45	109,4	3,13	20,39		
35,55	233,45	109,4	3,13	18,95		

Экономия $\delta_{ст}$ стального отражателя толщиной T находилась согласно экспериментально найденной зависимости

$$\delta_{ст} = 0{,}9T.$$

Указанная зависимость выполняется при $T \leqslant 1{,}0$ *см* для всех форм сборок в исследуемой области концентраций. Длина экстраполяции

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для параллелепипедов $d_{пар}$ не зависит в пределах экспериментальных ошибок от концентрации урана в растворе в исследуемой области (72 ÷ 233 *г/л*) и равна

$$d_{пар} = (2{,}65 \pm 0{,}15)\ см.$$

Таблица 2. Результаты критических экспериментов на сферах, содержащих растворы $UO_2(NO_3)_2$ 90%-ного обогащения

Без отражателя				С водяным отражателем			
Радиус сферы, см	Концентрация U^{235}, *г/л*	Отношение ρ_H/ρ_5	Отношение ρ_N/ρ_U	Радиус сферы, см	Концентрация U^{235}, *г/л*	Отношение ρ_H/ρ_5	Отношение ρ_N/ρ_U
20,1	43,2	600	2,5	20,1	32,4	800	2,5
19,0	50,6	500	2,5	19,0	35,5	720	2,5
18,1	67,0	380	2,5	18,1	37,5	680	2,5
17,17	92,8	264	3,3	16,4	50	500	2,5
16,4	118,3	208	2,5	16,0	54	465	2,5
16,14	151,5	156	3,2	13,5	106,4	233	2,5
16,0	194	119	3,15	12,7	172,8	139	2,5
16,0	202,5	118	2,5	12,7	182,0	127,5	3,28
16,0	287	86	2,5				

В табл. 2 приводятся результаты опытов на сферах без отражателя и с бесконечным водяным отражателем.

Длина экстраполяции сферического реактора $d_{сф}$ определяется из соотношения

$$d_{сф} = \frac{\pi}{B_{пар}} - R_0,$$

где $B^2_{пар}$ — лапласиан, определенный из опытов на параллелепипедах; R_0 — радиус сферы без отражателя. Длина экстраполяции для сферы в области концентраций 72 ÷ 233 *г/л* не зависит ни от концентрации урана в растворе, ни от кривизны поверхности

$$d_{сф} = (2{,}7 \pm 0{,}15)\ см.$$

Экономия отражателя из стали и воды определялась в специальных опытах на прямоугольном параллелепипеде, в котором толщина стали изменялась от 3 до 50 *мм*.

Экономия водяного отражателя для параллелепипедов $\delta_{пар}$ постоянна в пределах ошибок эксперимента для концентраций урана до 150 *г/л*

$$\delta_{пар} = (3{,}4 \pm 0{,}3)\ см.$$

Экономия водяного отражателя для сфер $\delta_{сф}$ в области концентраций урана 40 ÷ 230 *г/л* равна

$$\delta_{сф} = (3{,}4 \pm 0{,}3)\ см.$$

Основные результаты экспериментов приводятся на рис. 1. Эксперименты с раствором $UO_2(NO_3)_2$ при обогащениях урана 5 и 10% выполнены в цилиндрической геометрии (табл. 3). При расчете геометрических параметров растворов урана низкого обогащения используются длина экстраполяции и экономия водяного отражателя, полученные из экспериментов с растворами $UO_2(NO_3)_2$ 90%-ного обогащения. В данном случае ошибка в длине экстраполяции и экономии водяного отражателя, равная 8%, приводит к ошибке в геометрическом параметре не более 1%. Значения геометрических параметров, приведенные в табл. 2, рассчитаны с точностью около ±2%.

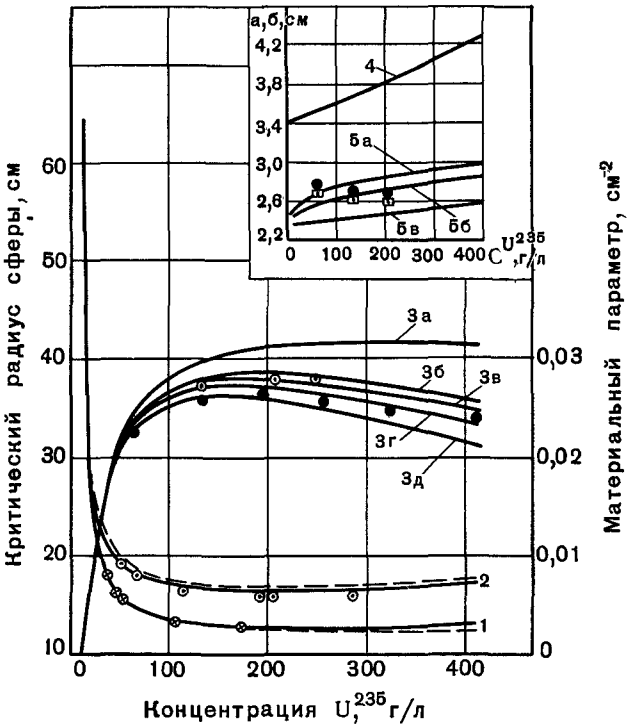


Рис. 1. Критические параметры водных растворов $U(90\%)O_2(NO_3)_2$: Отношение ρ_N/ρ_U = 2,5. Радиус сферы: *1* — с водяным отражателем; *2* — без отражателя; ○, ⊗ — эксперимент (сферы); ——— — расчет в многогрупповом P_3-приближении; - - - - — расчет в одногрупповом P_3-приближении. Материальный параметр: *3* — расчет по методу эмпирического ядра; *а* — из $\bar{K}(B)$[13]; *б* — $\bar{K}\left(\frac{B}{\sqrt{\rho}}\right)$; *в* — $\bar{K}\left(\frac{B}{\rho\ 0{,}6}\right)$; *г* — $\left(\frac{B}{\rho\ 0{,}75}\right)$; *д* — $\bar{K}(B/\rho)$; ⊙ — эксперимент; ⊕ — расчет по формуле $B^2 = \left(\frac{\pi}{R_3 + d_{сф}}\right)^2$; *4* — экономия водяного отражателя; *5* — длина экстраполяции: *а* — для сферы; *б* — для бесконечного цилиндра; *в* — для бесконечного плоского слоя. Эксперимент: ● — на сферах; □ — на параллелепипедах

На рис. 2 представлены лапласианы и радиус сфер с отражателем и без отражателя, полученные приравниванием лапласианов (см. табл. 3).

Теоретическое изучение критических параметров ядерных реакторов может осуществляться путем решения интегро-дифференциального уравнения Больцмана при соответствующем гра-

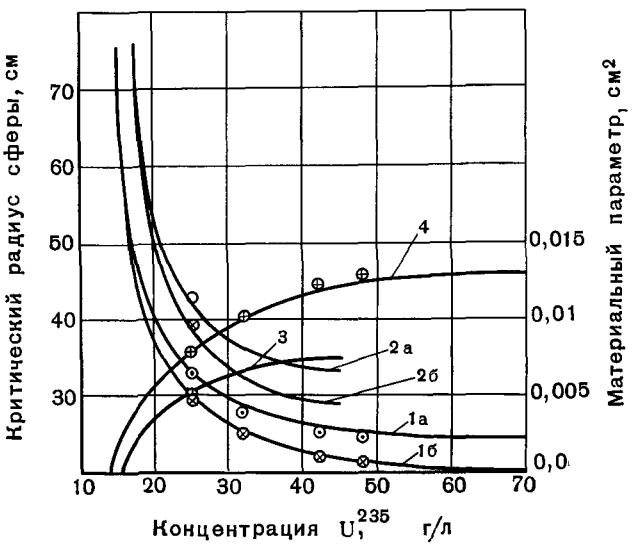


Рис. 2. Критические параметры водных растворов $UO_2(NO_3)_2$ (обогащения 5 и 10%). Отношение $\rho_N/\rho_U = 2{,}5$. Радиус сферы (расчет):

1 — 10%; *2* — 5% обогащения; *а* — сфера без отражателя; *б* — сфера с водяным отражателем. Материальный параметр по формуле $B^2 = \left(\frac{\pi}{R_э + d_{сф}}\right)^2$; *3* — 5%; *4* — 10% обогащения; ⊙, ⊗, ⊕ — из экспериментов на цилиндрах

ничном условии на внешней поверхности реактора.

В настоящем докладе для решения уравнения Больцмана будут применены многогрупповые методы расчета ядерных реакторов в P_1- и P_3-приближениях метода сферических гармоник [1] с использованием ЭВМ. В основу представленных исследований положены вопросы многогруппового представления и выбора многогрупповой системы констант [1, 2]. При расчете реакторов в P_1-приближении рассеяние в области замедления на всех ядрах, кроме водорода, учитывалось в диффузионно-возрастном приближении. Учет рассеяния на ядрах водорода производился точно. Многогрупповая система констант основывалась на данных о сечениях, приведенных в работах [3–5].

Учет резонансного поглощения производился в приближении узких резонансов. Для учета допплеровского уширения резонансов использовались данные работы [6]. Эффекты интерференции потенциального и резонансного рассеяний не учитывались.

Эффективные одногрупповые константы в области термализации получались усреднением по спектру нейтронов, который удовлетворяет уравнению Уилкинса [8]. Связь водорода в молекуле воды учитывается эффективной массой, равной ~ 6. Система констант для водорода и кислорода проверялась сравнением теоретических и экспериментальных значений квадрата длины замедления L_s^2 при энергии индиевого резонанса и длины диффузии тепловых нейтронов L (табл. 4).

Для сравнения теории и эксперимента по критическим массам в широкой области изменения концентраций необходимо вести расчеты в P_3-приближении. Уточнение критических размеров реактора за счет кинетических эффектов можно производить по эффективной одногрупповой теории [11].

Расчет критического радиуса сферы с водяным отражателем, содержащей растворы $UO_2(NO_3)_2$ 90%-ного обогащения, в многогрупповом P_3-приближении совпадает с экспериментом с точностью $\pm 1{,}5\%$. Расчет критического радиуса сферы без отражателя превышает результаты экспериментов на $2 \div 2{,}5\%$ (см. рис. 1). Аналогичные выводы получаются из сравнения эксперимента и расчета для растворов, содержащих уран 10%-ного обогащения (см. рис. 2). Единственный эксперимент для раствора урана 50%-ного обогащения дает значения критических радиусов сфер без отражателя и с отражателем, которые на 1% превышают значения радиусов, полученные в результате расчета в многогрупповом P_3-приближении. Таким образом, расчет реактора без отражателя в области высоких концентраций урана в растворе менее точен по сравнению с расчетом реактора с отражателем. Расчет кри-

Таблица 3. Результаты критических экспериментов на цилиндрах, содержащих раствор $UO_2(NO_3)_2$ 5%-ного и 10%-ного обогащения

Обогащение урана, %	Без отражателя						С водяным отражателем				
	Диаметр цилиндра, *см*	Концентрация U^{235}, *г/л*	Отношение ρ_N/ρ_U	Критическая высота, *см*	Длина экстраполяции, *см*	Геометрический параметр, *см*	Диаметр цилиндра, *см*	Концентрация U^{235}, *г/л*	Критическая высота H, *см*	Эффективная добавка, *см*	Геометрический параметр, *см²*
10	50	48,2	2,5	37,4	2,6	0,0128	40	48,2	36,8	6,0	0,0128
10	50	42,2	3,1	39,7	2,6	0,0123	50*	42,2	31,7	6,0	0,0122
10	50	32,2	3,0	54,1	2,6	0,0103	50*	32,4	40,7	6,0	0,0101
10	60	25,0	3,0	60,2	2,6	0,02767	50*	25,0	70,8	6,0	0,02763
5	80	25,3	2,4	70,4	2,6	0,02487	72,5	25,3	65,7	6,0	0,0249

* Верхний отражатель отсутствовал.

Таблица 4

Величина \ Источник	Эксперимент 8	Эксперимент 9	Эксперимент 17	Расчет
L_s^2	27,1±0,9	27,86±0,10	—	27,8
L	—	—	2,72	2,

тического радиуса сферы с отражателем в зависимости от концентрации урана 90%-ного обогащения в растворе при различных отношениях ядер азота и урана показывает, что минимальный радиус сферы с отражателем увеличивается при увеличении отношения ρ_N/ρ_U от 2 до 6,3 на 7 ÷ 8%. Сравнение результатов расчетов критических радиусов сфер по одногрупповой теории и в многогрупповом P_3-приближении показывает, что максимальное различие составляет 2,5 ÷ 3% и увеличивается с ростом концентрации урана в растворе.

Расчет материального параметра растворов $UO_2(NO_3)_2$ для урана 5, 10 и 90%-ного обогащения производится по формуле

$$B^2 = \left[\frac{\pi}{R_3 + d_{сф}}\right]^2,$$

где R_3 — критический радиус сферы без отражателя, рассчитанный в многогрупповом P_3-приближении; $d_{сф}$ — длина экстраполяции, полученная в P_1-приближении, равная $\frac{\pi}{B_1} - R_1$ (B_1^2 — материальный параметр среды в P_1-приближении; R_1 — критический радиус сферы без отражателя, рассчитанный в многогрупповом P_1-приближении с учетом энергетической зависимости длины экстраполяции). Расчет длин экстраполяции для бесконечного плоского слоя $d_{сл}$ и бесконечного цилиндра $d_ц$ производился в предположении, что длина линейной экстраполяции не зависит от формы активной зоны. Зависимость $d_{сф}$, $d_{сл}$ и $d_ц$ от концентрации U^{235} в растворе $UO_2(NO_3)_2$ показана на рис. 1. Длина экстраполяции увеличивается с ростом концентрации урана и увеличением кривизны поверхности активной зоны. Максимальное различие между расчетными и экспериментальными значениями длины экстраполяции для сферы составляет 7%. Максимальное различие экспериментальных и расчетных значений материального параметра для растворов 90%-ного обогащения в области концентраций урана 73 ÷ ÷ 233 *г/л* составляет 8%. Основное различие (7%) обусловлено расхождением расчетных и экспериментальных значений критических радиусов сфер без отражателя. Расхождение 7% в длине экстраполяции приводит к различию в лапласиане не более 1%. Для растворов урана 5%-ного и 10%-ного обогащения расхождение между расчетом и экспериментом составляет 6%, при этом 1% обусловлен разницей в длинах экстраполяции, а 5% — разницей в критических радиусах сферы без отражателя.

Экономия водяного отражателя для сферы находится как разность критических размеров без отражателя и с водяным отражателем. Расчет дает линейную зависимость между экономией водяного отражателя и концентрацией U^{235} в растворе (кривая *4* на рис. 1). Максимальное расхождение между расчетом и экспериментом составляет 10%.

К числу методов, основанных на измерениях макроскопических параметров реакторов, относится метод эмпирического ядра [12]. Метод эмпирического ядра применим к реакторам без отражателя, содержащим достаточно разбавленные растворы U^{235}, для которых выполняются следующие условия: 1) можно пренебречь делением на эпитепловых нейтронах; 2) размеры реактора достаточно велики, чтобы можно было пренебречь энергетической зависимостью длины экстраполяции. К системам, содержащим высококонцентрированные растворы урана 90%-ного обогащения, эти условия, строго говоря, неприменимы. Невыясненным является вопрос о влиянии молекул $UO_2(NO_3)_2$ на ядро замедления нейтронов в растворе. В настоящем докладе дается качественное объяснение этого вопроса. Предположим, что утечка нейтронов через поверхность реактора и резонансное поглощение нейтронов не коррелируют между собой. Тогда в соответствии с асимптотической теорией гомогенного реактора без отражателя можно записать следующее условие его критичности [12]:

$$\overline{K}(B;\ 1{,}4)\, e^{-B_2\Delta\tau_T}\left\{k_{\infty б} + \frac{k_{\infty т}}{1 + B^2L_т^2}\right\} = 1. \quad (1)$$

Предполагается, что основная доля делений нейтронов происходит в тепловой области. Величина $\overline{K}$ (B; 1,4) есть вероятность избежать утечки для нейтронов, достигающих энергии 1,4 *эв*. Функция $\overline{K}$ (B; 1,4) найдена М. Н. Ланцовым [13] путем обработки экспериментальных данных Хилла и др. [14] с поправкой на современное значение квадрата длины замедления нейтронов в воде. Входящие в формулу (1) параметры $k_{\infty б}$, $k_{\infty т}$ рассчитываются в многогрупповом P_1-приближении. Материальный параметр, найденный по формуле (1), показан на рис. 1 (кривая *3а*). Расчет по формуле (1) соответствует предположению, что замена молекул воды молекулами $UO_2(NO_3)_2$ не влияет на замедление нейтронов в растворе. Из сравнения расчетной кривой *3а* с экспериментальными данными по параллелепипедам видно, что такое представление не соответствует действительности. Запишем уравнение (1) в виде

$$K\left(\frac{B}{\rho\alpha};\ 1{,}4\right) e^{\frac{-B^2\Delta\tau_T}{\rho^2\alpha}}\left\{k_{\infty б} + \frac{k_{\infty т}}{1 + B^2L_т^2}\right\} = 1. \quad (1')$$

где ρ — эффективная плотность водорода в растворе;

α — коэффициент, учитывающий влияние молекул $UO_2(NO_3)_2$ на вероятность избежать утечки из реактора.

Сравнение с экспериментальными значениями геометрического параметра показывает, что α уменьшается с ростом концентрации урана в растворе. Это означает, что при возрастании концентрации урана увеличивается вклад замедления на молекулах $UO_2(NO_3)_2$. Расчет для 90%-ного обогащения показывает, что при одной и той же концентрации урана материальный параметр уменьшается при увеличении отношения ρ_N/ρ_U, причем на 80% за счет уменьшения плотности водорода в растворе и на 20% за счет поглощения азотом. Для растворов урана низкого обогащения значение поглощения азотом увеличивается.

2. КОМПЕНСИРУЮЩАЯ СПОСОБНОСТЬ ГЕТЕРОГЕННЫХ ПОГЛОТИТЕЛЕЙ НЕЙТРОНОВ В ВОДНЫХ РАСТВОРАХ $UO_2(NO_3)_2$ 10%-НОГО И 90%-НОГО ОБОГАЩЕНИЯ

Эксперименты по определению компенсирующей способности гетерогенных поглотителей нейтронов проводились в цилиндрических сборках диаметрами 400 и 1100 *мм*. Изучалось изменение критической высоты раствора уранилнитрата от введения в активную зону сборок цилиндрических и кольцевых вставок, системы и решеток стержней.

Поглощающим материалом служили листовой кадмий толщиной ~0,5 *мм* и порошок карбида бора плотностью $\gamma = 1{,}25$ *г/см*^{3}; поглотители нейтронов заключались в чехлы из стали 1Х18Н9Т. Эксперименты проводились в условиях, описанных в первом разделе доклада. Точность измерений критических высот раствора составляет $\pm(1{,}5 \div 2)$ %.

Рассмотрим результаты экспериментов и расчетов по определению компенсирующей способности гетерогенных поглотителей нейтронов в сборке диаметром 400 *мм*.

Компенсирующая способность центрального поглощающего стержня зависит от материала, диаметра стержня и концентрации урана в растворе. В эксперименте чехлы из стали переменной толщины, содержащие поглощающий материал, вводились в раствор на всю высоту сборки. На рис. 3 показаны эксперименты со стержнями из карбида бора. Стальная оболочка приводит к двум противоположным эффектам: экранирует карбид бора и вытесняет раствор. Как видно из рис. 3, эффект вытеснения преобладает. Расчет компенсирующей способности центрального стержня из карбида бора производится по одномерной программе в многогрупповом P_1-приближении на ЭВМ, эффективное граничное условие на поверхности стержня зависит от альбедо стержня с оболочкой, которое рассчитывается по методике Е. И. Гришанина [15]. Расчет согласуется с экспериментом (см. рис. 3) в пределах $10 \div 25$%, причем согласие улучшается при увеличении диаметра стрежня.

Рассмотрим компенсирующую способность стержней из карбида бора и стержней из кадмия с водяным замедлителем. Как показывает эксперимент (рис. 4), при концентрации урана 136 *г/л* до диаметра, составляющего приблизительно две длины замедления нейтронов, стержень из карбида бора обладает большой компенсирующей способностью по сравнению со стержнем из кадмия с водой. При диаметрах, превышающих две длины замедления нейтронов, стержень из кадмия с водой оказывается эффективнее стержня из карбида бора. При увеличении размера стержня из карбида бора увеличивается его самоэкранирование, а при увеличении размеров стержня из кадмия и воды увеличивается доля быстрых нейтронов, которые замедляются в стержне ниже порога поглощения кадмия.

Рассмотрим компенсирующую способность системы поглощающих стержней (1 в центре и 6 равномерно по кольцу). Компенсирующая способность системы стержней из карбида бора ϕ 58/ϕ50 *мм* и из кадмия с водой ϕ 57/ϕ 46 *мм* имеет интерференционный максимум (рис. 5) при некотором расстоянии между центрами стержней. Оно зависит от концентрации урана в растворе, типа стержней и присутствия отражателя нейтронов. На рис. 6 дана зависимость критической высоты раствора в сборке без отражателя от среднего радиуса кольцевой вставки. Вставки имели толщину зазора 6 *мм*, толщину оболочки 1 *мм*. Поглощающий материал — карбид бора и кадмий. Рис. 6 показывает, что наибольшая компенсирующая способность наблюдается при среднем радиусе вставки, составляющем ~0,45 R (R — радиус активной зоны). Эксперименты на сборке с отражателем дают оптимальный размер вставки ~0,53 R. Кольцевые вставки из карбида бора эффективнее вставок с кадмием при всех радиусах вставок.

Компенсирующая способность вставки с двумя коаксиальными кадмиевыми цилиндрами растет при заполнении ее водой с увеличением толщины слоя воды (рис. 7).

В экспериментах на сборке диаметром 110 *см* определяется компенсирующая способность решеток стержней диаметром 45 *мм* по B_4C с толщиной стальной оболочки 5 *мм*. Установка и фиксация стержней осуществляются двумя стальными трубными решетками толщиной 17 *мм* (одна на дне активной зоны, другая на верхнем фланце корпуса). Диаметр отверстий в трубных решетках 58 *мм*. Результаты экспериментов представлены в табл. 5. Анализ компенсирующей способности решеток поглотителей производится методом эквивалентной цилиндризованной ячейки с эффективным границ-

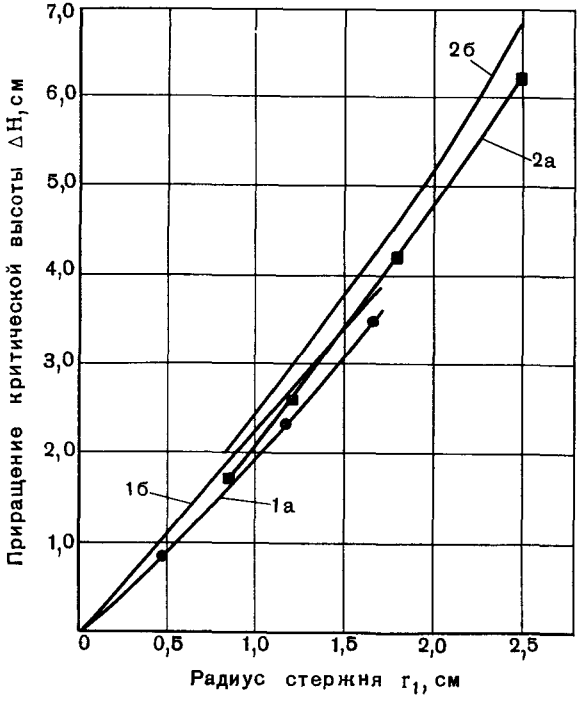


Рис. 3. Зависимость приращения критической высоты раствора от радиуса стержня (по карбиду бора). Диаметр сборки без отражателя 40 см. Концентрация урана 72 г/л. Обогащение 90%:

1 — толщина стальной оболочки 0,03—0,05 *см*; *2* — толщина стальной оболочки 0,4 *см*; *а* — эксперимент; *б* — расчет

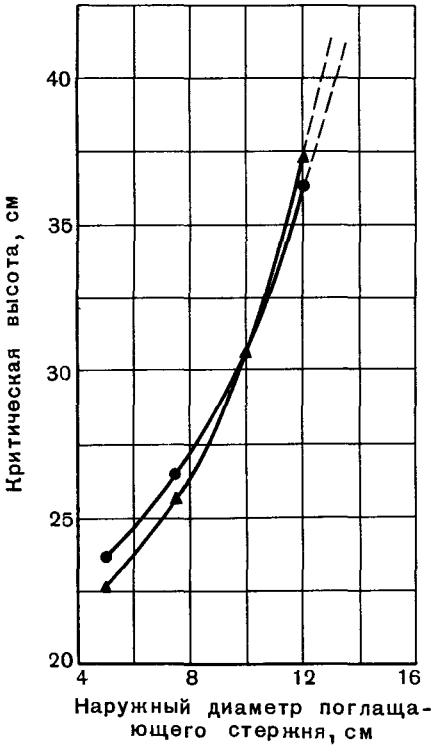


Рис. 4. Зависимость критической высоты от диаметра поглощающего стержня Сборка диаметром 40 см без отражателя. Концентрация урана 136 г/л. Толщина стальной оболочки вставки 0,1 см:

▲ — кадмиевый стержень с водой, толщина кадмия 0,05 *см*; ● — стержень из карбида бора плотностью 1,25 *г / см³*

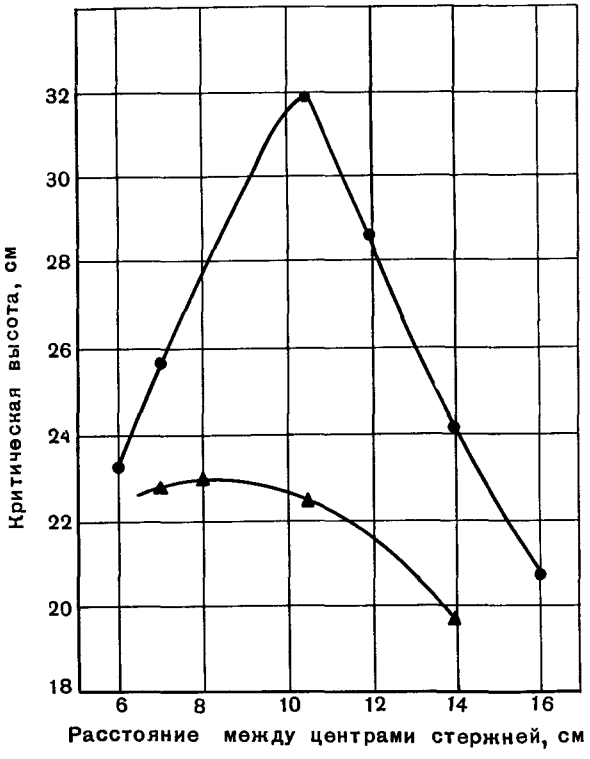


Рис. 5. Зависимость эффективности системы из семи поглощающих стержней от расстояния между их центрами. Сборка диаметром 40 см с боковым и нижним водяными отражателями:

● — стержни из карбида бора в трубах ⌀ 5,8×0,4 *см*, C_U = 289 *г / л*; ▲ — кадмиевые стержни с водой ⌀ 4,6×0,05 *см* в трубах ⌀ 5,7×0,55 *см*, C_U = 457 *г / л*

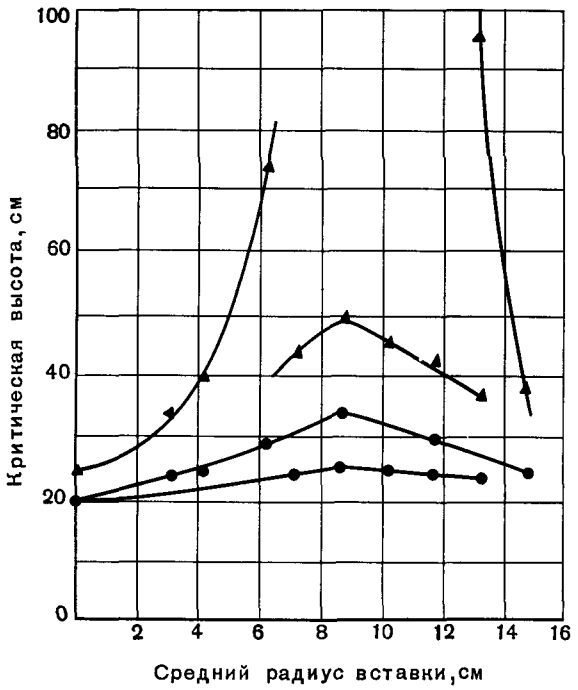


Рис. 6. Зависимость критической высоты раствора от среднего радиуса кольцевой поглощающей вставки. Сборка диаметром 40 см без отражателя. Толщина оболочки вставки 0,1 см. Обогащение 90%:

1 — поглотитель — карбид бора; толщина кольцевого зазора 0,6 *см*; *2* — поглотитель — кадмий толщиной 0,05 *см*; ▲ — $C_U = 76$ *г/л*; ● — $C_U = 289$ *г/л*

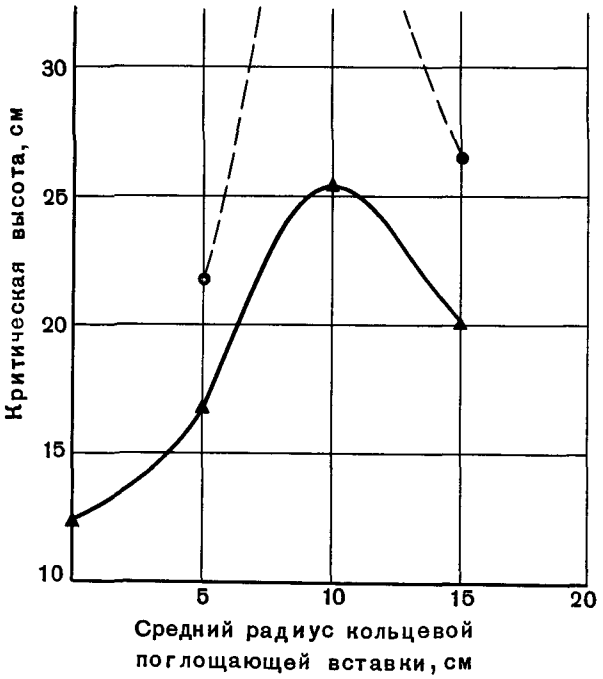


Рис. 7. Зависимость критической высоты раствора от среднего радиуса кольцевой вставки с водой. Сборка диаметром 40 см с боковым и нижним отражателями. Толщина стальной оболочки вставки 0,1 см, толщина кадмия 0,05 см. Концентрация урана 289 г/л. Обогащение 90%. Расстояние между двумя коаксиальными кадмиевыми цилиндрами:

▲ — 2,5 *см*; ● — 5,0 *см*

ным условием на поверхности стержня. В случае «тесного» расположения поглощающих стержней в решетке учитывается их взаимная экранировка. Поглощающие свойства стержня в ячейке могут быть охарактеризованы эффективным макроскопическим сечением $\Sigma_{эфф}(U)$, которое

Таблица 5. Критические высоты раствора $UO_2(NO_3)_2$ в сборке диаметром 110 см с водяным нижним и боковым отражателями при наличии в активной зоне сборки компенсирующей решетки поглощающих стержней ∅ 55 × 5 мм

№ п./п.	Обогащение U^{235}, %	Концентрация урана в растворе, *г/л*	Шаг треугольной решетки, *см*	Количество стержней в решетке, шт.	Критическая высота раствора, *см*		$\frac{H_{расч} - H_{эксп}}{H_{эксп}}$, %
					Расчет $H_{расч}$	Эксперимент $H_{эксп}$	
1	5,64	400,24	31,8	7	—	103,8	—
2	10	420,46	22,8	19	—	27,434	—
3	10	420,46	18,4	31	—	34,164	—
4	90	37,51	13,4	51	—	102	—
5	90	37,51	22,8	19	23,3	23,0	+1,5
6	90	37,51	15,2	41	38,8	40	—3
7	90	37,51	7,6	41	26,1	24,7	+6
8	90	74,87	21,2	19	15,6	17,0	—8
9	90	74,87	15,2	40	20,3	20	+1,5
10	90	74,87	13,2	51	23,4	24,2	—3,0
11*	90	152,3	21,2	19	13,2	13,3	+1
12*	90	152,3	7,6	61	14,6	14,45	+1

* Опыты проводились с поднятой нижней трубной решеткой.

зависит от состава стержня, шага решетки и энергии нейтронов. Энергетическая зависимость $\Sigma_{эфф}(U)$ может быть получена методом групп [16] и в диффузионно-возрастном приближении. Диффузионно-возрастной метод получения $\Sigma_{эфф}(U)$ основан на предположении, что плотность замедления нейтронов можно представить в виде

$$q(\rho, U) = R_U(\rho) e^{-\int_0^u \frac{\Sigma_c}{\xi\Sigma_s} du' - \int_0^u \frac{\varkappa^2 u D}{\xi\Sigma_s} dU'},$$

причем основная зависимость от энергии заключается в экспоненте. Функция $R_U(\rho)$, описывающая пространственное распределение плотности замедления, удовлетворяет уравнению

$$\Delta R_U(\rho) + \varkappa_U^2 R_U(\rho) = 0$$

с соответствующими граничными условиями [16]. В табл. 5 приводятся результаты расчета в P_1-приближении критических высот реактора диаметром 110 *см* с треугольными решетками стержней ϕ 45/ϕ 55 *мм*. Расчет $\Sigma_{эфф}(U)$ выполнен в диффузионно-возрастном приближении с учетом взаимной экранировки стержней в «тесной» решетке. Зависимость диффузионных и замедляющих свойств решетки стержней от шага решетки

не учитывается. Это должно приводить к занижению критических расчетных высот по сравнению с экспериментальными. Сравнение расчета и эксперимента показывает, что диффузионно-возрастная методика расчета $\Sigma_{эфф}(U)$ завышает компенсирующую способность «тесных» решеток стержней и занижает компенсирующую способность «редких» решеток.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/381 USSR

The critical parameters of aqueous solutions of $UO_2(NO_3)_2$ and nuclear safety

By B. G. Dubovsky *et al.*

Experimental and theoretical studies of the critical parameters of aqueous solutions of $UO_2(NO_3)_2$ with 90, 10 and 5% uranium enrichment are described. These studies were carried out with the object of determining nuclear safety conditions and studying homogeneous reactor physics. The critical volumes of the solutions for various uranium concentrations in the range 30-450 g/l were determined.

Extrapolation distances and the savings resulting from the use of various reflectors were determined during the experiments.

The experimental results obtained for cylindrical and parallelepipedal cores by transformation of the geometric buckling using the extrapolation distance and reflector saving of a light-water reflector were recalculated for cores in the shape of a sphere, an infinite cylinder and a slab of infinite area, and the results compared with those of experiments carried out on cores of the first two shapes.

The paper includes the results of experiments for determining: (i) the dependence of the critical size on the percentage of steel and nitric acid in the core, and (ii) the effect of various compound reflectors on the critical sizes.

The experimental data were subjected to theoretical analysis in two ways: by direct solution of multi-group equations for a reactor in P_1 and P_3 approximations and by the empirical kernel method. An idealised problem of a water-moderated reactor was solved to check the accuracy of the various approximations. It is shown that use of the calculation and cross-section method gives results that agree well with the experimental data.

The material buckling and extrapolated boundary of water-moderated reactors are derived from this analysis.

The paper also gives the results of experiments carried out to determine the effectiveness of absorbing rods of different shapes and compositions in homogeneous uranium-water reactors containing uranyl nitrate solutions.

The effectiveness of regular absorbing-rod lattices of various pitches and various uranium concentrations in the solution was also determined.

Multi-group calculations for determining the effectiveness of a central cylindrical rod of boron carbide with a steel sheath are also reported.

The calculation method put forward in the paper for determining the effectiveness of a regular absorbing-rod lattice in a reactor consists in homogenising the core with control rods.

The effectiveness and homogenisation parameters of infinite absorbing-rod lattices in homogeneous multiplying media are determined, allowance being made for the detailed dependence on the energy of the absorption and scattering cross-sections of the rod material.

The effectiveness and cros sections of infinite lattice rods are derived in a diffusion-age approximation on the basis of the analysis of an equivalent cell in one-dimensional geometry.

The calculated absorption cross section with homogenisation of an absorbing rod in a cell was taken as the basis for multi-group calculation of the effectiveness of rod lattices, totally or partly filling the core, in a homogeneous reactor.

The results of experimental studies and theoretical calculations on neutron interaction of subcritical assemblies with aqueous solutions of uranyl nitrate are presented.

A/381 URSS

Paramètres critiques des solutions aqueuses de $UO_2(NO_3)_2$ et sûreté nucléaire

par B. G. Dubovsky *et al.*

Le mémoire décrit une étude expérimentale et théorique des paramètres critiques de solutions aqueuses de $UO_2(NO_3)_2$, l'uranium étant enrichi à 90, 10 et 5%. Cette étude a été entreprise pour déterminer les conditions de sûreté nucléaire et étudier la physique des réacteurs homogènes. On a déterminé les volumes critiques des solutions pour différentes concentrations de l'uranium allant de 30 à 450 g/l.

Les expériences ont permis de déterminer les longueurs extrapolées et l'économie due à l'emploi de différents réflecteurs.

Les résultats expérimentaux obtenus pour des cœurs cylindriques et parallélépipédiques, par transformation du laplacien géométrique, en utilisant les longueurs extrapolées et l'économie due à l'emploi de l'eau comme réflecteur, ont été recalculés pour des cœurs ayant la forme d'une sphère, d'un cyclindre infini et d'une plaque infinie.

Ces valeurs ont été comparées aux résultats des expériences faites sur des cœurs en forme de sphère et de cylindre infini.

On donne les résultats d'expériences tendant à déterminer i) la relation entre les dimensions critiques et le pourcentage d'acier et d'acide nitrique contenus dans le cœur et ii) l'influence de différents réflecteurs composés sur les dimensions critiques.

L'analyse théorique des données expérimentales a été faite de deux manières: résolution directe des équations multigroupes du réacteur par les approximations P_1 et P_3 d'une part et par la méthode des noyaux empiriques d'autre part. Pour vérifier la précision des différentes approximations, on a procédé à la résolution d'un problème idéalisé de réacteur ralenti à l'eau. On montre que la méthode de calcul et les sections efficaces utilisées permettent de décrire les résultats des expériences de manière satisfaisante.

Sur la base de l'analyse effectuée, on tire des conclusions concernant le laplacien matière et la limite extrapolée des réacteurs ralentis à l'eau.

Le mémoire donne les résultats d'expériences tendant à déterminer la capacité compensatrice de barres absorbantes de forme et composition diverses dans les réacteurs homogènes uranium-eau avec des solutions de nitrate d'uranyle.

On détermine la capacité compensatrice des réseaux réguliers de barres absorbantes pour divers pas de réseau et différentes concentrations de l'uranium dans la solution.

On donne les calculs multigroupes relatifs à la capacité compensatrice d'une barre cylindrique centrale en carbure de bore gainée d'acier.

La méthode de calcul proposée dans le mémoire pour déterminer la capacité compensatrice d'un réseau régulier de barres absorbantes dans le réacteur consiste en l'homogénéisation du cœur avec barres absorbantes.

La capacité compensatrice et les paramètres d'homogénéisation d'un réseau infini de barres absorbantes dans des milieux multiplicateurs homogènes sont déterminés en tenant compte de la relation détaillée entre l'énergie et les sections efficaces d'absorption et de diffusion du matériau des barres.

La capacité de compensation, de même que les paramètres d'absorption des barres d'un réseau infini, sont déterminés par l'approximation diffusion-âge sur la base de l'analyse d'une cellule équivalente dans la géométrie unidimensionnelle.

Sur la base du calcul de la section efficace d'absorption avec homogénéisation de la barre absorbante dans une cellule, on a établi une méthode multigroupe de détermination de la capacité compensatrice d'un réseau de barres absorbantes dans un réacteur homogène remplissant en partie ou totalement la zone active du réacteur.

Le mémoire donne les résultats expérimentaux et les calculs concernant l'étude de l'interaction neutronique d'assemblages sous-critiques avec des solutions aqueuses de nitrate d'uranyle.

A/381 URSS

Parámetros críticos de las soluciones acuosas de $UO_2(NO_3)_2$ y seguridad nuclear

por B. G. Dubovsky *et al.*

La memoria describe un estudio experimental y teórico de los parámetros críticos de las soluciones acuosas de $UO_2(NO_3)_2$, con uranio enriquecido al 90, 10 y 5%. El estudio se ha realizado para determinar las condiciones de seguridad nuclear y estudiar la física de los reactores homogéneos. Se han determinado los volúmenes críticos de las soluciones para diversas concentraciones de uranio comprendidas en el intervalo 30-450 g/l.

Los experimentos han permitido determinar las longitudes de extrapolación y el ahorro debido a la utilización de diversos reflectores.

Los resultados experimentales obtenidos para los núcleos cilíndricos y paralelepipédicos, mediante la transformación de la laplaciana geométrica, utilizando las longitudes de extrapolación y del ahorro debido a la utilización del agua como reflector, se han calculado de nuevo para los núcleos de forma de esfera, de cilindro infinito y de placa infinita.

Se han comparado estos valores con los resultados de experimentos realizados con núcleos de forma de esfera y de cilindro infinito.

Se dan los resultados de los experimentos realizados para determinar i) la relación que existe entre las dimensiones críticas y la proporción de acero y de ácido nítrico en el núcleo, y ii) la influencia de los diferentes reflectores compuestos sobre las dimensiones críticas.

El análisis teórico de los datos experimentales se ha efectuado por dos métodos: por la resolución directa de las ecuaciones del reactor con varios grupos en las aproximaciones P_1 y P_3, y por aplicación del método de los núcleos empíricos. Para comprobar la precisión de las distintas aproximaciones se resolvió el problema idealizado de un reactor moderado con agua. Se demuestra que el método de cálculo y las secciones eficaces utilizadas permiten expresar los resultados experimentales en forma satisfactoria.

Basándose en el análisis efectuado se deducen las laplacianas materiales y los límites extrapolados de los reactores moderados con agua.

La memoria da los resultados de los experimentos realizados para determinar la capacidad de compensación de barras de absorción de formas y composiciones diversas, en reactores homogéneos de uranio-agua con soluciones de nitrato de uranilo.

Se ha determinado la capacidad de compensación de redes regulares de barras de absorción para distintos pasos de red y distintas concentraciones de uranio en la solución.

Se han realizado cálculos, con varios grupos, para determinar la capacidad de compensación de una barra central, cilíndrica, de carburo de boro con vaina de acero.

El método de cálculo propuesto en la memoria para la determinación de la capacidad de compensación de una red regular de barras de absorción en un reactor consiste en la homogeneización del núcleo con barras absorbentes.

La capacidad de compensación y los parámetros de homogeneización de una red infinita de barras absorbentes en medios multiplicativos homogéneos se determinan teniendo en cuenta la relación de dependencia detallada entre la energía y las secciones eficaces de absorción y de dispersión del material de las barras.

La capacidad de compensación y los parámetros de absorción de las barras de una red infinita se determinan por la aproximación de la edad-difusión, sobre la base del análisis de la celda equivalente en geometría unidimensional.

Basándose en el cálculo de la sección eficaz de absorción con homogeneización de la barra absorbente en la celda se ha elaborado un método con varios grupos para determinar la capacidad de compensación de una red de barras absorbentes en un reactor homogéneo, que ocupa parcial o totalmente el núcleo del reactor.

La memoria presenta los resultados experimentales y los cálculos concernientes al estudio de la interacción neutrónica de conjuntos subcríticos con soluciones acuosas de nitrato de uranilo.

Experience in the transport of irradiated fuel

By N. F. C. Bishop,* J. C. Egan * and W. W. Marshall **

The Production Group of the UKAEA has been concerned for the past 10 years with the transport of radioactive materials, both within and outside its establishments. Many of the movements have taken place so often that they are now routine and in the case of irradiated Magnox fuels large quantities are regularly moved.

The movements described in this paper are confined to those of large quantities of irradiated fuel moved on a routine basis between establishments in the United Kingdom and to the more recent extension of the service to overseas establishments. The development work now in hand to extend further the overseas service is also briefly considered.

SOME ESTABLISHED MOVEMENTS WITHIN THE UNITED KINGDOM

For convenience in presentation, the various movements are described under the heading of the type of container used.

It is important to state at this point that except for the CEGB flask the containers were designed and commissioned before formal approval procedures were instituted and in fact formal approval is not yet required for road movements in the United Kingdom. Some flasks may, therefore, require to be moved on the basis of special arrangements when the United Kingdom Statutory Regulations first come into force.

MTR flask

This flask (Fig. 1) has been used since 1956 for the transport of irradiated fuel elements from the DIDO and the PLUTO materials testing reactors (MTR) at Harwell to the Chemical Reprocessing Plant at Dounreay.

First movements were by rail and these continued until 1962, when British Railways became concerned that the impact strenght of the flask might not be adequate. Movements were then continued by road while the impact problem was re-examined. To satisfy British Railways' requirements it was necessary to strengthen the holding-down arrangements on the wagon and to demonstrate by calculation that the wagon frame would absorb some of the energy in an impact accident, and that the flask would remain attached to the wagon. Rail movements were then restarted in 1963, subject to a maximum speed of 40 mph.

This cube-shaped flask of approximately 4 feet side is fabricated in cast iron, lead and steel; it weighs approximately 15 tons and has capacity for 25 MTR type fuel elements. The maximum ^{235}U fuel load determined by criticality considerations is 4.6 kg. Fission product heat output is estimated to be 2 kW, the bulk of which is generated in the shields, a smaller part being transferred from the fuel elements to the shield by conduction and air convection. The total heat is dissipated to the atmosphere outside the flask by natural convection from the surface. Fuel and flask temperatures are within those specified by the IAEA regulations. The flask has been shown to be capable of withstanding a free fall of 12 feet on to a hard surface.

The Certificate of Approval for transport packaging design issued by the UKAEA Health and Safety Branch approves a package comprising the MTR flask attached to a specific conveyance in a specified manner. Transfer of the flask to and from such conveyance is subject to the terms of a special arrangement. A Criticality Clearance Certificate under Fissile Class II is given against the terms of the above packaging certificate.

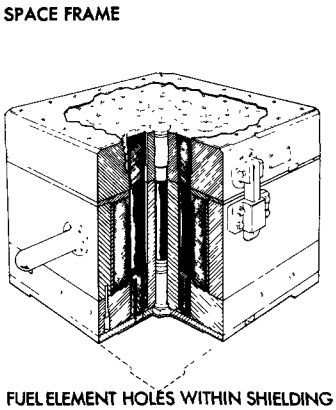


Figure 1. MTR flask

* U.K. Atomic Energy Authority, Production Group, Risley.
** U.K. Atomic Energy Authority, Production Group, Windscale.

A similar Australian MTR flask was also used for the transport of fuel from Australia to the UK and, in order to qualify for criticality clearance under IAEA Fissile Class I, was enclosed in a 4 inches thick wooden crate reinforced with steel. This crate also affords protection against impact and, because of the ablative properties of the wood, protection against fire.

However, the low impact resistance of the cast iron flask remains an embarrassment and work is proceeding on the design of an MTR flask in fabricated steel.

Chapelcross flask

The flask has undertaken more than a thousand journeys without a radiation incident; vehicle breakdowns and accidents have occurred, including one head-on collision with a lorry loaded with railway axles, but in no case was the flask itself damaged.

The flask (Fig. 2) is used almost exclusively for conveying irradiated Magnox-canned natural uranium fuel by road from Chapelcross to Windscale, a distance of about 70 miles. The route is through Carlisle, a town of 71 000 population, and Egremont, a village with narrow streets. The route was selected in conjunction with the Ministry of Transport, the Police and local authorities concerned, as one which would ensure minimum inconvenience to other road users.

The flask weighs about 40 tons and in normal use carries 140 elements in a steel skip immersed in water. Fission-product heat is transferred through the water to the flask walls and dissipated to atmosphere by natural convention from the surface. Temperature and radiation levels are within those specified by the IAEA regulations. However the flask, which is made of cast iron, does not at the moment qualify for a Type B safety approval, and therefore is one of those mentioned as possibly requiring special arrangements under future UK regulations.

Under normal transport conditions, hydrogen evolved by Magnox corrosion is retained in the flask and the pressure rise in the 35 litres of ullage does not exceed 5 psi compared with the lid seal test pressure of 20 psi. However, if the flask were seriously delayed due to an accident, then an emergency team including health physics specialists would be sent to the scene of the accident to measure the flask temperature and if necessary to vent the flask through a trap system. So far this procedure has not had to be invoked.

Central Electricity Generating Board flask

The flow of irradiated fuel elements from the Bradwell and Berkeley power stations has commenced, and as further stations are commissioned, the flow will increase to a rate of two flasks per day over the next decade.

The CEGB is responsible for the transport of irradiated Magnox-canned natural uranium fuel from its reactors to the reprocessing plant at Windscale. Flasks are transported by road to the nearest rail head and then by rail to Windscale, the average journey time

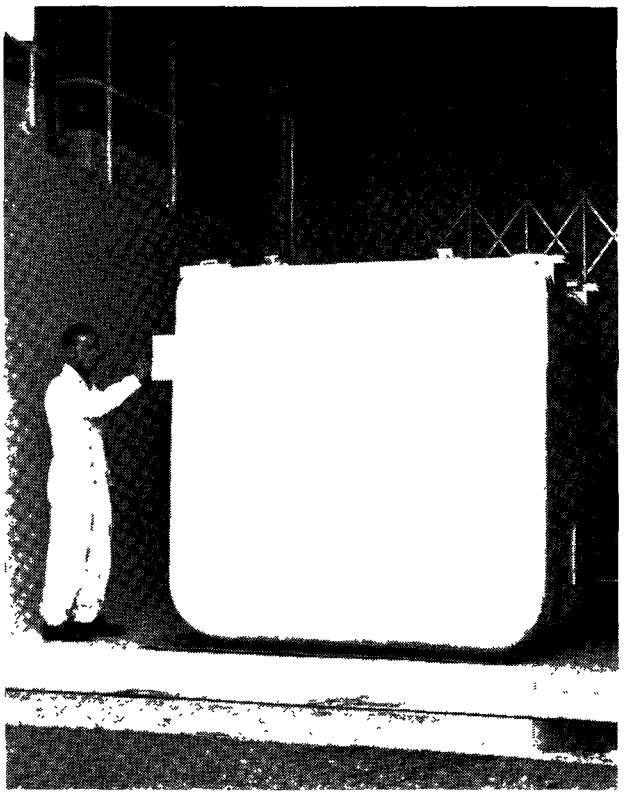


Figure 2. Chapelcross flask

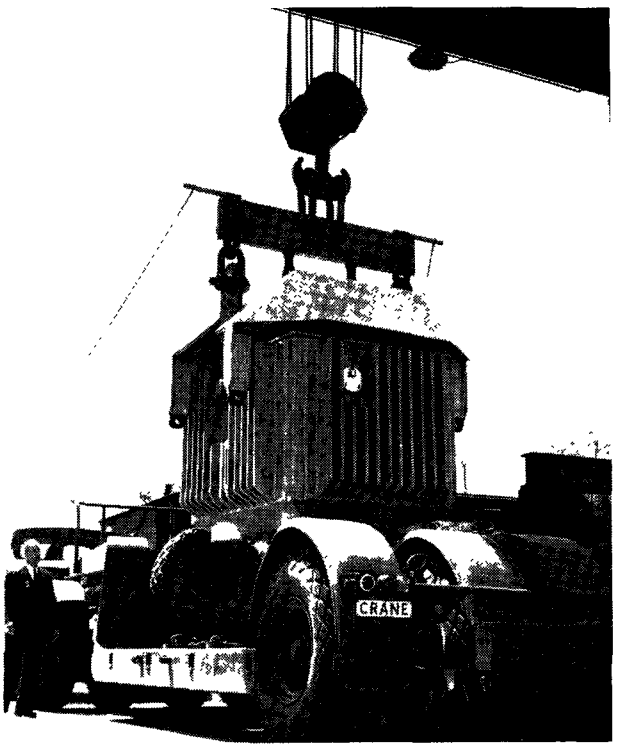


Figure 3. CEGB flask

being 3 to 5 days. The flask wagons travel as part of normal express goods trains and, except for the presence of a British Railways' inspector, there are no special arrangements. An emergency procedure has however been agreed between British Railways, CEGB and UKAEA so that, in the event of an accident, emergency teams will be directed to the scene of the accident.

A 50 ton fabricated steel flask (Fig. 3) developed from a UKAEA design for cast iron or cast steel [1] is used in conjunction with a specially designed railway wagon provided by British Railways. The flask has capacity for about 2 tonnes of fuel in a skip immersed in water. The decision to use water was based, in the first instance, upon the handling techniques provided both at the civil reactor sites and at Windscale and it is also the most convenient way of transferring the fission product decay heat to the walls of the flask — the external surface of the flask is cooled by natural air convection. The effect of solar heat is reduced by the provision of a shield over the top surface of the flask. Irradiated fuel is cooled for approximately 100 days before despatch and, contingent upon the previous irradiation history, the maximum fission-product heat may be 8.5 kW giving a bulk water temperature of between 70 and 80 °C depending upon solar heating and an external flask surface temperature of 60 to 70 °C. These temperatures are within the requirements of the IAEA regulations.

When the flask contains 2 tonnes of fuel irradiated at 3 MW/t to 3 000 MWd/t, hydrogen generated by Magnox corrosion and radiolysis produces a daily increase in pressure of 3 to 6 psi. Before dispatch the flask is purged with nitrogen at atmospheric pressure. The nitrogen-hydrogen mixture is retained in the flask during the journey and released in a ventilated cell on arrival at Windscale.

Magazine flask

The flask (Fig. 4) is a combination of discharge machine and magazine type transport flask and was designed for the discharge and transport by road, of up to six short-cooled Magnox-canned natural uranium fuel elements, from reactors to the examination or experimental facilities in Authority establishments. It is used when it is essential to keep the fuel dry — usually Magnox fuels are cooled in water at reactor sites before dispatch for reprocessing. The flask has been used for the transport of fuel between Chapelcross and Windscale and Harwell. In both cases the route has been agreed with Borough and County surveyors and about one week is required to alert the authorities before each movement. Except for the usual radiation and contamination measurements before dispatch and the instructions carried by the driver, no special precautions are taken, except those to minimise interference with other road users, and this is normal practice for heavy loads.

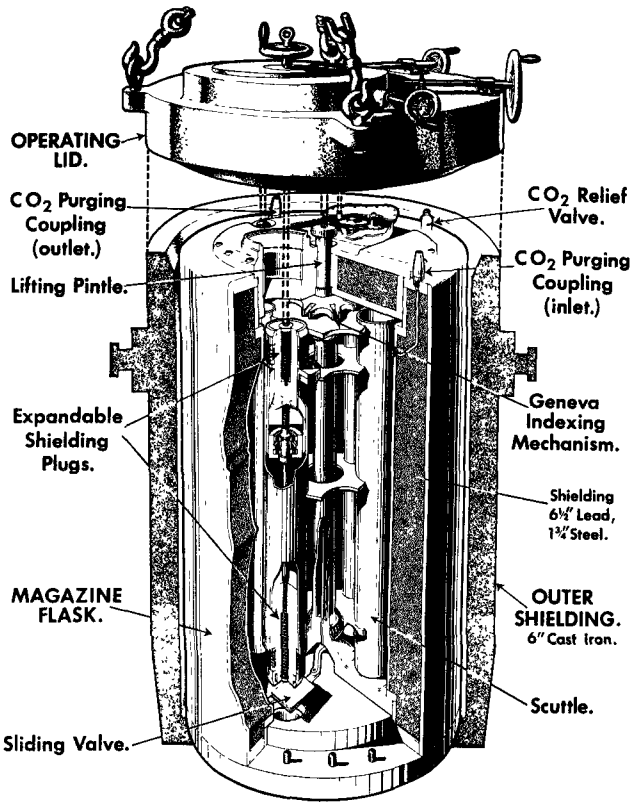


Figure 4. Magazine flask with outer shielding

The flask comprises two concentric cylindrical containers weighing 20 tons. The shielding of the inner container comprises $6\frac{1}{2}$ inches of lead and $1\frac{3}{4}$ inches steel, the lead being sandwiched between the steel. The outer container is made of 6 inches of cast iron. The six fuel elements are lifted into the magazine by a winch mounted on top of the flask, and rotating and indexing mechanisms are provided.

RECENT OVERSEAS MOVEMENTS

During the last two years irradiated fuel from both the Danish DR3 reactor and the Australian HIFAR MTR has been reprocessed at Dounreay. In both cases the fuel was transported by sea and rail using MTR flasks of the types described earlier.

Australia-United Kingdom

Irradiated elements from the HIFAR reactor of the Australian Atomic Energy Commission at Lucas Heights were transported by sea from Melbourne to Liverpool via Cape Town and from Liverpool to Dounreay by rail. Six AAEC flasks to Australian design developed from a UKAEA design and fitted with detachable wooden crates reinforced with steel were used for the operation. Each flask carried 25 fuel elements containing approximately 2 kg of ^{235}U.

The AAEC chartered the M.V. *Nella Dan*, a 2 000 ton general cargo ship of the Lauritgen Line, which left Melbourne on 12 April 1963 and arrived at Liverpool on 5 June. The flasks were carried in the hold, immediately below the fore-hatch.

At Liverpool, the consignment was taken in charge by UKAEA personnel and the flasks transferred by floating crane to six large flat railway wagons and conveyed by special goods train to Thurso for Dounreay — the railway wagons are the same as those used for CEGB flasks.

No special precautions were taken other than to ensure that the emergency procedures normally provided for the rail movements of irradiated fuel in the UK were available; these procedures have already been described.

Denmark-United Kingdom

During the period September 1962 to April 1963, four flasks of irradiated fuel were returned from the Danish DR3 reactor. Each consignment comprised approximately 2 kg ^{235}U contained in 25 fuel elements to UKAEA design.

All these shipments have been made at normal commercial rates from Copenhagen to Leith in the 1 000 ton Currie Line ship M.V. *Finland*. The first flask was transported from Leith to Dounreay by road on a UKAEA vehicle, but subsequent movements were made by rail.

EXPANSION OF TRANSPORT SERVICES

The transport of the MTR fuel from overseas establishments to UKAEA reprocessing plants was regarded by the Authority as the beginning of a new phase of transport. The Authority accordingly sought to expand these activities by undertaking further movements or by placing their experience, technical and engineering skills at the service of customers. Reactor operators have shown an increasing interest in this service and many transport problems have been examined. Some of these are discussed below.

Magnox fuels

A gas-cooled reactor using Magnox-canned natural uranium fuel is in operation in Italy (Latina) and a similar reactor is being built in Japan (Tokai Mura). UKAEA personnel have prepared technical, safety and economic assessments for the transport of irradiated fuel from these reactors to the reprocessing plant at Windscale in the UK. In both cases there is a considerable financial incentive to use water-filled flasks of the type used by the CEGB and it was therefore necessary to demonstrate that the long transport times involved would not adversely effect the condition of the fuel or the safety of the movement. The relevant development work is described later.

Oxide fuels

Oxide fuels, compared with Magnox fuels, differ widely in both nuclear and constructional details and, of course, large quantities have not been transported commercially within the United Kingdom. However, from a transport point of view, these fuels have several common features, for example:

(*a*) In most cases the materials of construction do not corrode at temperatures permitted by transport regulations;

(*b*) Fission-product heat release is comparatively intense;

(*c*) Fuel is usually enriched and some form of criticality safety limitation may be imposed;

(*d*) Fuel pins are assembled in clusters or stringers of up to 12–15 ft long (4–5 meters) and reactor operators often prefer the transportation of complete stringers.

At present the most attractive proposition is for a flask capable of servicing several reactors and carried on a special vehicle similar to that shown in Fig. 5. It has been established that this type of equipment can be made to satisfy both Continental and United Kingdom rail requirements without limitations on speed, and could also be carried by sea. However, although technically feasible the weight of such a flask in steel (60–90 tons) may be too heavy for existing handling equipment at some reactor sites and a lighter flask carrying fuel broken into short lengths may be a more acceptable alternative.

DEVELOPMENT

The development of irradiated fuel containers is centred on the selection of materials of construction, the experimental verification of shielding and heat-transfer calculations and the problems associated with the corrosion of canning materials and radiolysis of heat-transfer media.

Materials of construction

Hitherto cast iron and lead (contained in steel) have been used extensively in the fabrication of irradiated fuel flasks. No operational difficulties have arisen from their use although serious objections have been raised by safety organisations and transport authorities: it is claimed that the low impact strength of cast iron may result in a fracture which could permit the release of activity in impact accidents and that the low melting point and large thermal expansion of lead may result in reduction of shielding or possibly loss of containment in an accident in which a flask is involved in a

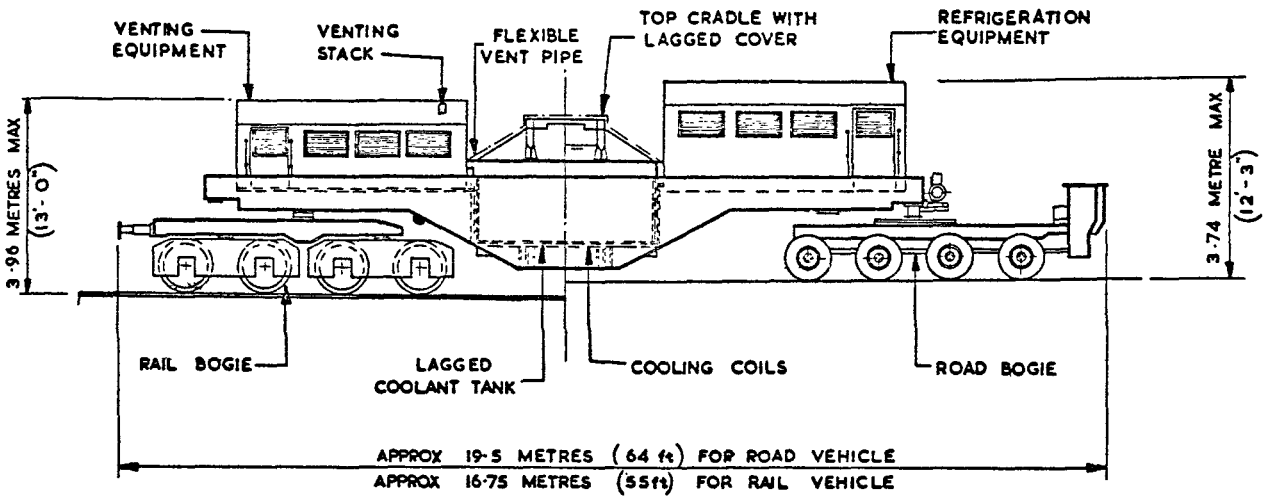


Figure 5. Proposed design of road/rail vehicle

fire. These are justifiable objections but they need not be regarded as precluding further examination, by designers, of methods of using these materials to produce flasks which will satisfy the performance requirements of competent authorities and transport operators. Cast iron in particular should not be discounted since malleable forms are available and the fabrication cost compared with equivalent fabricated steel containers is lower. In certain cases the risk of melting of lead in a fire can be considerably reduced by the use of ablative materials such as wood, a case in point being that of the flask used for the transport of MTR fuel from Australia to the United Kingdom.

Work is in progress on the examination of the use of concrete as a shielding materials in flasks. Although the use of concrete increases the bulk of containers, this tendency can be markedly reduced by the use of depleted uranium oxide as aggregate.

Heat transfer

In view of the high capital cost of irradiated fuel flasks, it is necessary to predict the temperature conditions of the fuel, heat transfer medium and flask in advance of manufacture. However, in many cases, particularly when fuel elements are randomly packed as in the case of the Chapelcross and CEGB flasks or in static gas as for the magazine flask, estimated temperatures may vary considerably from those met in practice. With this in mind, designers of earlier flasks used conservative heat-transfer assumptions.

To optimise this aspect of flask design it is necessary to check calculated temperature conditions by experiment. The principal difficulty with such experiments is in the measurement of the surface temperature of irradiated fuel unless the elements have been specially prepared before loading into the reactor. Furthermore, the use of irradiated fuel in heat transfer experiments considerably increases the difficulties of experimental techniques. However, in many cases, close approximation to true conditions can be achieved by the use of dummy fuel elements containing electrical heaters.

CEGB have determined heat transfer data for the civil fuel flask by heating the flask water with immersion heaters and simulating solar heat with lamps. The calculated conditions were confirmed.

In the case of the MTR flask, fuel temperature measurements were made at Harwell with thermocoupoles in the thermocouple pockets of irradiated fuel elements and on the lead and external surface of the flask. In this case also good correlation was obtained between the calculated and measured temperatures.

Corrosion and radiolysis

The internal pressure in flasks may increase in transit due to relative expansion of the flask and heat-transfer medium and the gaseous products of radiolysis and/or corrosion. The effect of volume changes in the flask and medium are readily assessed from existing information, but further experimental work is often required to obtain suitable radiolysis and corrosion data.

The corrosion of Magnox-canned fuels in water was studied jointly by the Consortia, the CEGB and the UKAEA using inactive specimens which had been coated with an oxide-graphite layer representative of that produced on magnox under reactor conditions. The principal conclusion was that corrosion could be minimised by using demineralised water and maintaining a pH of 11.5 by the addition of 200 ppm of caustic soda — the chloride content of the water must be less than 1 ppm as chloride. The work also demonstrated that the corrosion rate depends upon the condition of the Magnox surface, that is, the composition and thickness of the carbon-oxide layer, and the temperature of the surface; typical corrosion rates measured during these experiments were 1 mg/dm^2 day at 60 °C and 12.5 mg/dm^2 day at 100 °C.

Because the data obtained from such experiments can be misleading, particularly when pitting corrosion takes place, the UKAEA have undertaken full scale experiments using irradiated fuel. In one of these experiments the water in the flask was maintained at about 10 °C (to minimise corrosion) for a period of about three weeks to assess the effect of long transit times on the condition of the fuel; the failure of cooling equipment during transit was also studied. The results, which are shown in Table 1, provided the basis for the proposal to use refrigerated flasks for the transport of Magnox fuel from Italy to the United Kingdom.

To establish the corrosion component of hydrogen generation, it was necessary to estimate the effect of radiolysis. Irradiation experiments using suitably treated water in the presence of Magnox indicated a G_{H2} value of 0.35. Estimates of the percentage of absorption range from 5 to 10%, depending upon the theoretical model adopted. Measurements of gamma flux by dosimetry in the experimental flask and in trials with single elements confirmed the above values.

A preliminary examination was also made of alternatives to water as heat-transfer media. This work shows that the use of organic liquids, notably tricresylphosphate, appears to overcome the corrosion problem if their water content is below 0.5%. However moisture contents of this order are difficult to achieve and maintain at reactor and reprocessing sites where water shielding is used during fuel-transfer operations.

COSTS

Capital

Approximate capital costs for the containers described in this report are:

Chapelcross flask .	£ 3 000—£ 5 000
CEGB flask . . .	£12 000—£15 000
MTR flask . . .	£ 5 000—£ 7 000

The principal reason for the high cost of the CEGB flask is the use of fabricated steel and the expensive jointing techniques adopted. As previously mentioned the Chapelcross flask is cast iron and the MTR flask cast iron and lead. The estimated cost of a CEGB flask in cast iron is £6 000–£8 000.

In addition to flasks it may be necessary to provide special road vehicles, railway wagons and handling equipment. In the case of sea journeys, modifications to ships may be necessary; the following estimates give some indication of the possible magnitude of these costs:

(*a*) The road-rail wagon for a 50 ton flask carrying refrigeration equipment capable of dissipating 10 kW of fission heat would cost approximately £40 000;

(*b*) A road vehicle of the type used for the Chapelcross and CEGB flasks costs about £15 000;

(*c*) A rail wagon for an air-cooled CEGB type flask costs £10 000 to £12 000.

(*d*) Modifications to a ship to carry say six 50 ton flasks and including the provision of 30 kW of refrigeration capacity would cost £20 000–£30 000.

Haulage charges

The following are examples of haulage charges made for movements within and to the UK:

Type of flask	Transport by	Number of flasks	Cost/mile £
MTR	Rail	1	0.8
MTR	Road	1	0.6
MTR	Charter ship	6	0.7[a]
CEGB	Rail	1	0.9
Chapelcross	Road	1	0.5

[a] The charge quoted is that for the Australia-United Kingdom voyage and may have been low because the ship was in any event returning empty to the United Kingdom.

At present no firm quotations have been received from railway authorities on the Continent of Europe. However, available information on standard freight rates indicates that they may be higher than those quoted above.

Estimates have been obtained for various sea movements which indicate that the freight charge for a 50 ton flask will be approximately £1 per mile when carried

Table 1. Results of corrosion trials

Total cooling time (days)	Location of fuel	Temperature (°C)	Hydrogen evolved (litres/day)		
			Total	Radiolysis (calculated)	Corrosion (by difference)
0–116	Chapelcross pond	12–22	No reliable values		
116–119	Flask	22–10	Transport and decreasing temperature		
119–143	Static flask	10	6	2	4
143–145	Static flask	10–50	No values—Increasing temperature		
145–150	Static flask	50	16	2	14
150–157	Static flask	22	8	2	6
157–177	Static flask	12	5.5	2	5.3

on large ships (say 10 000–15 000 tons) as part cargo; for some journeys lower freight charges may be charged when a ship of between 1 000–2 000 tons is chartered to carry a number of flasks for a period of several months in each year. A typical division of transport cost is as follows:

		% Total
Haulage—30 journeys of 600 miles at £0.8/mile	£14 400	81
Depreciation—£22 000 for flask and wagon over 10 years	2 200	12
Interest (say 6%)	660	4
Maintenance, say	440	3
Total annual cost	£17 700	100

The haulage rate includes insurance against normal accidents; indemnity costs are not included.

Clearly, the dominant factor is haulage cost and as this is usually related to the total weight or volume of the load, there is a strong incentive to increase the payload-to-total-weight ratio. However, shielding and impact strength are controlled by regulatory requirements and therefore an increase in this ratio could be achieved only by an increase in over-all dimensions; there are of course practicable limitations to the maximum size of flask which can be handled and transported. Further, an increase in payload may necessitate the use of mechanical cooling systems to dissipate fission-product heat generated in flasks and this will also increase both the capital and operating costs.

Rail transport generally is operated by national organisations and reduction in cost may be achieved by negotiation as confidence in the movement of nuclear materials increases. In the case of transport by road, it is possible for members of the nuclear industry to provide their own equipment. However, unless full utilisation is achieved, reductions in haulage costs will not accrue. Similarly, in the case of sea transport, ownership or the use of charter agreements may result in lower costs, although the former would require considerable capital.

THE FUTURE

In future more reactors will be built and operated in countries which, for various reasons prefer not to build expensive reprocessing plants similar to those available at Windscale. The operators of such reactors will be interested in arranging for their irradiated fuel to be processed in existing plants; thus the number of international movements of irradiated fuel will increase.

The cost of establishing and operating an international transport service is considerable even when data are available from other nuclear activities, as is the case in the UKAEA.

The Production Group of the UKAEA are aware of this situation and have, in fact, already offered transport services to reactor operators as part of reprocessing agreements by undertaking all or part of movements of irradiated fuel to the United Kingdom.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/184 Royaume-Uni

Expérience dans le transport de combustible irradié

par N. F. C. Bishop *et al.*

On décrit les mouvements normaux de combustible irradié dans le Royaume-Uni; cela comprend le transport des combustibles irradiés en uranium naturel à gainage de Magnox par route de Chapelcross à Windscale et par fer des centrales nucléaires civiles à Windscale. On fournit des détails sur les containers de transport, les véhicules et l'organisation générale de ces mouvements. On inclut également une description d'un container dans lequel on transporte les éléments secs à gainage de Magnox.

On donne des renseignements similaires pour le transport des éléments combustibles du type MTR (réacteur d'essai de matériaux), à la fois entre Harwell et Dounreay et d'Australie et du Danemark à Dounreay.

On discute l'extension de ce service de transport à d'autres réacteurs d'outre-mer, eu égard notamment au travail d'étude technique en vue du transport de combustible à Magnox irradié d'Italie et du Japon aux installations de traitement en Grande-Bretagne. Cela comprend une description du type de véhicule qui pourrait être utilisé à la fois pour transporter des combustibles à Magnox dans de l'eau à basse température pour éviter les effets de la corrosion et aussi des combustibles provenant de réacteurs améliorés modérés à l'eau et à réfrigérant gazeux, pour lesquels un équipement mécanique de refroidissement est nécessaire pour satisfaire les exigences réglementaires en matière de température.

On donne un aperçu du travail d'étude et de mise au point; on décrit en particulier le travail expérimental

sur le transport des combustibles gainés au Magnox dans un milieu aqueux de transfert de chaleur, qui a abouti à la décision d'utiliser des containers de transport à réfrigération mécanique. On discute le choix et l'utilisation d'autres moyens de transfert de chaleur et on donne une indication des problèmes soulevés par l'utilisation de fluides organiques.

On fournit des indications sur le coût des containers et des véhicules de transport ainsi que le tarif du fret par fer et par mer.

On montre que, comme la construction de l'équipement de transport est en relation étroite avec les exigences réglementaires, il ne sera sans doute pas possible d'effectuer des réductions substantielles dans les frais d'investissements nécessaires pour cet équipement, mais que des réductions dans le coût général des transports devraient pouvoir être obtenues en maintenant des facteurs élevés d'utilisation de l'équipement. On émet l'avis que la tendance suivant laquelle l'exploitant d'un réacteur accepte de se charger du retour du combustible irradié n'est pas économique et qu'il faudrait encourager la création d'un nombre limité d'organisations spécialisées de transport.

A/184 Соединенное Королевство

Эксперимент по транспортировке облученного топлива

Н. Ф. К. Бишоп *et al.*

Описывается установившийся порядок перевозки облученного топлива в пределах Великобритании, включая перевозку облученных тепловыделяющих элементов из естественного урана в магноксовой оболочке по дороге из Чепел-кросса в Виндскейл и по железной дороге от гражданской атомной электростанции Соединенного Королевства в Виндскейл. Обсуждаются подробности, касающиеся транспортных контейнеров и средств перевозки, а также общая организация перевозок. Дается описание контейнера для перевозки тепловыделяющих элементов с магноксовой оболочкой.

Приводится подобная же информация по транспортировке тепловыделяющих элементов реактора типа MTR как между Харуэллом и Даунри, так и из Австралии и Дании в Даунри.

Обсуждается дальнейшее развитие транспортного обслуживания реакторов, строящихся за границей, с подробным изложением инженерно-технических вопросов транспортировки облученных тепловыделяющихся элементов в магноксовой оболочке из Италии и Японии на перерабатывающие радиохимические заводы Великобритании. Включено описание типов транспортных средств перевозки, которые могут быть использованы как для транспортировки магноксовых тепловыделяющих элементов в воде при низкой температуре, что сводит до минимума коррозионные эффекты, так и топлива усовершенствованных реакторов с газовым охлаждением и водяным замедлителем, требующего принудительного охлаждения оборудования для поддержания соответствующих температурных условий.

Приведен перечень проектных и опытно-конструкторских работ, включающий описание экспериментов по транспортировке тепловыделяющих элементов в магноксовой оболочке в водной среде с теплоотводом, которые привели к использованию транспортных контейнеров с принудительным охлаждением. Обсуждаются вопросы выбора и использования теплопередающей среды, и указываются некоторые проблемы, возникающие при использовании органических жидкостей.

Приводится стоимость транспортных контейнеров, средств перевозки, а также стоимость перевозки как железнодорожным, так и морским транспортом.

Показано, что поскольку на конструкцию транспортного оборудования налагаются жесткие требования, то нельзя добиться существенного снижения капитальных затрат при изготовлении этого оборудования. Однако снижение общей стоимости транспортных операций может быть достигнуто в результате повышения коэффициента использования оборудования. Предполагается, что современная тенденция возлагать на владельцев реакторов ответственность за переработку облученного топлива является экономически нецелесообразной и что должен поощряться рост ограниченного числа специализированных организаций по перевозке облученного топлива.

A/184 Reino Unido

Experiencia adquirida en el transporte de combustibles irradiados

por N. F. C. Bishop *et al.*

La memoria describe los desplazamientos habituales de combustible irradiado dentro del Reino Unido; entre ellos se encuentra el transporte por carretera de combustible de uranio natural revestido con Magnox desde Chapelcross a Windscale y el transporte por ferrocarril desde las centrales nucleoeléctricas del Reino Unido hasta Windscale. Se detallan los recipientes utilizados, los vehículos y la organización general de los transportes: también se incluye la descripción de un recipiente para el transporte de elementos secos revestidos con Magnox.

Se suministran datos similares sobre el transporte de elementos combustibles del tipo MTR entre Harwell y Dounreay y desde Australia y Dinamarca a Dounreay.

Se discute la ampliación de este servicio de transporte a otros reactores de ultramar, principalmente en relación con la asistencia técnica para transporte de combustible Magnox irradiado desde Italia y el Japón a las instalaciones de tratamiento del Reino Unido. A este respecto se describe el tipo de vehículo que podría servir para el transporte de combustibles Magnox en agua, a baja temperatura a fin de reducir a un mínimo los efectos de corrosión, y combustibles procedentes de reactores avanzados refrigerados por gas y reactores moderados por agua, para los que se requiere un sistema mecánico de refrigeración con regulación de temperatura.

Se reseña el plan básico y el desenvolvimiento de las tareas realizadas, con una descripción de los trabajos experimentales sobre el transporte de combustible Magnox en agentes acuosos de transmisión de calor, de donde surgió la decisión de emplear recipientes de transporte refrigerados mecánicamente. Se discute la selección y el empleo de diversos medios de transmisión de calor y se indican los problemas que plantean los medios orgánicos.

Se indican también en la memoria los costes de los recipientes y vehículos y las tarifas de transporte por ferrocarril y por mar.

Por último se observa que no parece posible una reducción importante en el coste del equipo de transporte porque su construcción está sujeta a severas especificaciones legales, pero los gastos totales de funcionamiento pueden aminorarse si se mantiene elevado el factor de utilización del equipo. Por otra parte, la tendencia actual de que el propio explotador del reactor devuelva el combustible irradiado es antieconómica, por lo que se sugiere la conveniencia de establecer un número limitado de organizaciones especializadas en transportes.

Safety aspects of spent fuel transport

By J. W. Langhaar,* A. E. Aikens, Jr.,** N. J. Rigstad,*** H. E. Walchli **** and L. L. Zahn, Jr.*****

During the past ten years the shipment of kilocurie and megacurie quantities of radioactive material, particularly in the form of spent fuel from nuclear reactors, has become rather common. Although safety has been a primary consideration in all cask designs, definition of the hazards and adoption of a uniform degree of safety have become increasingly important for continued protection of the public and for economic development of the industry. The two principal documents describing current US practice and views are the Atomic Energy Commission's proposed regulation 10 CFR Part 72 [1] and the American Standards Association's proposed guide [2].

Of principal concern is exposure of the public to radioactive material, which, if released, could be damaging to property and harmful to personnel. Containers for spent fuel, commonly called casks in the US, are therefore designed so that the risk of release is negligible under normal conditions of operation and handling. The major areas of current investigation relate to defining conditions of casks malfunction or accident sufficiently likely or severe to justify their serving as a design basis. It must also be determined whether or not a particular cask would function as intended under adverse conditions. Since such risks cannot be reduced to zero, criteria and designs must be developed which represent a degree of safety compatible with accepted practice for other hazardous materials, taking into account the various aspects of personal injury and overall cost.

RECENT UNITED STATES EXPERIENCE

During the period 1957–1963, three minor rail accidents, two minor truck accidents, and one moderately severe truck accident involved shipments of spent fuel [3]. In no case was the cask subjected to severe impact or fire, nor was the cask seriously damaged. This favorable experience does not indicate any deficiency of design, but it is only a small sample and thus does not serve as a suitable basis for prediction.

On four other occasions, contaminated water escaped from casks as a result of inadequate design and testing rather than vehicle accident. These occurrences did not result in personal injury, but have directed greater attention to the design of cask closures and vent systems.

The general cargo accident rate in the US is about 3 per million vehicle miles for accidents resulting in vehicle damage of more than $100 for trucks or $2 000 for trains [4, 5]. Although complete data are lacking, it appears that the frequency rate during 1957–1963 for shipments or radioactive material was somewhat smaller than for general cargo [6*a*]. While the number of cask shipments and miles traveled during this period are not known, it seems unlikely that the frequency rate for spent fuel differed greatly from that for other radioactive materials.

CHARACTERISTICS OF FUEL

The outstanding feature of spent fuel, compared with more familiar forms of radioactive material, is its high specific radioactivity. This is evidenced by the high levels of penetrating radiation and the high rates of heat evolution. In some cases precautions must be taken to avoid a possible nuclear chain reaction, and often the chemical and mechanical properties of fuel assemblies require special consideration.

Fuels from research and test reactors have commonly been highly enriched U-Al fuels with 20 to 30% burn-up, with decay periods of 90 to 120 days, and with decay heat of several hundred watts per assembly.

Power reactor fuels have usually been slighlty enriched UO_2 pellets clad either with stainless steel or Zircaloy. Peak burn-up has exceeded 20 000 MWd per tonne of total uranium, and may exceed 30 000 MWd per tonne in the future. Some fuel assemblies in current use contain more than 200 kg of uranium and, during reactor operation, produce more than 5 MW each. A single assembly at the time of shipment may contain more than 10^6 curies of activity and generate 5 to 10 kW of heat. Several such assemblies may be shipped in a single cask. It is expected that most shipments will

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***** Hanford Atomic Products Operation, Richland, Washington.

be to reprocessing sites. Prior to reprocessing, at least 150 days of cooling will be required. Sufficient fuel for a reprocessing campaign can therefore be accumulated with shipments starting 90 days or more after reactor discharge.

Fuel assemblies at the time of shipment, if unshielded, would sometimes produce a gamma radiation field exceeding 10 000 r/h at 1 m. The shielding requirement varies with type and quantity of fuel, but is customarily in the range of 7 to 9 in of lead or equivalent for compliance with shipping regulations.

Knowledge of decay heat is important for cask design. Unfortunately, present data do not permit accurate calculation. Some uncertainties in fission yields, decay schemes, and average energy of beta disintegrations exist, such that values of decay heat after several months of cooling, as calculated from data in various reports, differ by as much as $\pm 20\%$. Further inaccuracy arises from inexact knowledge of burn-up distribution throughout a reactor, and from minor effects of such variables as moderator temperature, fuel temperature, neutron flux, and neutron energy. Thus, since accurate prediction of fuel temperatures during transport is impracticable, it is common practice to neglect (*a*) the escape of beta and gamma energy from the fuel, (*b*) the axial flow of heat in fuel assemblies, and (*c*) the heat capacity of fuel and cask as a mechanism for smoothing out peaks.

Fuels of different design and composition also vary greatly in nuclear reactivity. Fuels of up to about 2% enrichment normally do not require special precautions against criticality with quantities of one cask load. By contrast, highly enriched fuel of the MTR plate type, with 140 to 200 g of ^{235}U per element, has a minimum critical mass of about 2 500 g of ^{235}U, based on optimum spacing in natural water [7]. Twenty-mil cadium sheets clad with stainless steel, and 250-mil boral plates have been used effectively as nuclear poisons in casks. The boral has been found to have a thermal neutron attenuation exceeding 99.9% [8]. For fast criticality, the minimum critical mass of ^{235}U is 22.8 kg [9]. A quantity of 10.0 kg in a cask is thus considered safe under possible accident conditions involving loss of coolant and melting of fuel.

The effects of air, water, and other coolants on cladding, bonding, and fuel core materials; the diffusivity of uranium with certain cladding materials; and the effect of temperature on the physical properties of the materials must be taken into account in cask design. Extensive information of this nature is presented in *Reactor Handbook*, Vol. 1, *Materials*, 2nd ed.

Data regarding the escape of fission products from fuel at high temperatures are useful for estimating fuel failure temperature and for assessing the hazards of certain accident conditions. Although such data are not available for all types of fuel, experiments with irradiated U-Al have shown almost no release of nonvolatile fission products at temperatures below the melting point of aluminium [10]. At slightly higher temperatures, some release occurs as the melt solidifies. In the absence of other failure mechanisms such as chemical reaction or mechanical dispersal, 660 °C is thus taken as the failure temperature for this type of fuel. With stainless-steel-clad UO_2 fuel, the estimated failure temperature is commonly based on rupture of the cladding from internal pressure, with suitable experimental basis for the assumed amount of free gas.

HANDLING METHODS

The customary fuel handling systems at reactor sites provide a means to remove spent fuel from the reactor and place it in storage racks in a waterfilled pool that provides shielding and cooling. Small fuel assemblies may be stored and shipped in re-usable canisters. The pools are equipped with light cranes and tools for moving fuel and depositing it in the shipping cask. Casks are set on the bottom of a section of the pool, at water depths of 20 ft or more as required for shielding, by means of an overhead bridge crane or a field crane. Casks for research fuel commonly weigh from 10 to 20 tons, single element casks for power reactor fuel from 20 to 30 tons, and multi-element casks from 30 to 100 tons.

Fuel assemblies or canisters are loaded one at a time and in such a way as to avoid the possible formation of a critical array. The cask lid is installed under water, and the cask is then lifted by the crane, usually using rigid yokes rather than slings. A yoke may have open hooks that engage trunnions on the cask, or trunnions that engage sockets on the cask.

Safety bolts are usually installed in the cask lid while it is under water or while the cask is suspended partly immersed. Final closure bolts are installed after cask removal from the pool. An area adjacent to the pool is provided for cask decontamination and for sampling the coolant. After determination that the cask is ready for shipment, it is lifted to the transport vehicle by crane and attached to the vehicle. Height restrictions on transport necessitate shipping long casks in a horizontal position and may also require special methods for tilting the cask. Tilting can be accomplished with two overhead cranes, or with a single crane and a fixture for pivoting the cask.

Handling methods at receiving sites are similar. Recently designed receiving sites at the Savannah River Plant and at Nuclear Fuel Services, Inc., are further equipped with a vent system for safe relief of internal pressure, with means for safe disposal of contaminated coolant, and with facilities for internal cooling of dry casks with steam and/or nitrogen before immersing them in water. Special storage and handling

methods are used for fuels that are highly reactive with water, such as those with sodium bonding or those containing uranium carbide [11, 12].

PERFORMANCE REQUIREMENTS

Aside from shielding against direct radiation, the primary intent of all precautions in design and transport of casks is to prevent uncontrolled release of radioactive material to the environment. In addition to causing personal injury, wide dispersal of radioactive material could cause large economic loss. Such release would be associated with damage to the cask or its contents as a result of fire, impact, loss of heat dissipation, or a criticality incident. Any of these could result directly or indirectly from a transportation mishap. Consequently, a basic approach to safe shipment is to provide safeguards against these mechanisms of failure. Another perhaps equally important approach is to reduce the frequency and severity of mishaps by the use of administrative safeguards.

For practical reasons, cask performance requirements are stated indirectly in terms of resistance to certain environmental conditions and in terms of temperature, pressure, and radiation level. It is necessary to define accident conditions which relate to performance requirements. Proposed regulation 10 CFR Part 72 postulates a 15-ft drop onto an unyielding surface, and a penetration type of impact plus a standard 1-h fire test as two independent events. The ASA proposes a 30-ft drop, followed by a fire of as yet unspecified severity. The difference is due to the fact that selection of the justifiable degree of risk must be somewhat arbitrary in the absence of suitable data on actual accidents. Consequently, the AEC has studies under way to provide more information about the expected frequency and severity of accidents.

Radiation levels for casks have been based on other previously established limits considered to provide reasonable assurance of no personal injury and no damage to other nearby materials such as photographic film. Current limits required by the AEC are 200 mr/h at the surface and 10 mr/h at 3 m from the surface of an exclusive use vehicle. One of the ICC limits is 10 mr/h at 1 m from the source; however, special authorizations are available.

Under accident conditions, AEC regulations permit up to 1 r/h at the cask surface. The ASA has further recommended considering 1 r/h as the allowable average over the surface, with local levels up to 100 r/h. This is because even in the absence of immediate control measures, individuals would not be expected to remain very long in a high radiation zone in such an emergency. In the case of lead-shielded casks, rupture of the outer shell or failure of lid fastenings is regarded as *prima facie* evidence of excessive external radiation levels, and must therefore be avoided.

Radiation exposure of personnel handling these casks is also a design consideration. Since a single crew may regularly handle casks, limits set by reprocessing sites are about 25 man-millirems per cask for the unloading operation. This limit is readily met with small casks, but the longer exposure time for large multi-element casks may necessitate added shielding or special remote handling tools.

A temperature of 180 °F at the accessible surface of the cask is considered safe for personnel. Internal temperatures are governed by physical properties of materials involved, and also by the potential failure of fuel at high temperature. In normal operation, a fuel temperature either at least 300 °F below failure temperature or no higher than during normal reactor operation, with suitable adjustment for any important effect of difference in external pressure on the fuel, is considered acceptable. Under accident conditions, 10 CFR Part 72 permits 200 °F below failure temperature. The ASA suggests further consideration of stating the requirement in this case in terms of expected release of radioactive material from the cask. For all such evaluations, factors to be considered in addition to fission gases include internal pressure, creep characteristics of fuel and cladding, and possible gas or moisture remaining in the fuel after fabrication.

Excessive release of radioactive material could result from an accidental nuclear chain reaction. The quantity of fissile material, spacing and arrangement of fuel, neutron moderation and reflection, presence of neutron absorbers, and neutron interaction with other fissile materials which may be in close proximity must be taken into account. An effective neutron multiplication constant (k_{eff}) less than unity represents subcriticality; however, because of uncertainties in the calculations and the possible occurrence of more unfavorable conditions than assumed for the calculations, the specified maximum k_{eff} is 0.90. The assumed conditions include (*a*) optimum moderation and reflection of neutrons by water in and around the cask, since actual distribution of water in accidents and even in normal handling is unpredictable; (*b*) a degree of irradiation of the fuel corresponding to maximum reactivity, since there may be inaccuracy in burn-up determinations and error in fuel identification; (*c*) the most reactive fuel array permitted by the cask dimensions and structure under postulated accident conditions; and (*d*) the minimum predictable effectiveness of nuclear poisons under postulated accident conditions. Since the effectiveness of such poisons may be adversely affected by mechanical damage or by chemical and physical processes, proposed regulations require suitable periodic tests. Structural integrity requirements for the cask are intended to provide adequate assurance that fuel will not escape from the cask; therefore the criticality control criteria do not consider the possibility of rearrangement into a more reactive array

outside the cask. Reliance is placed on administrative procedural control during normal loading and unloading operations.

Criticality resulting from neutron interaction with other containers of radioactive material, either on the same vehicle or at handling sites, can be avoided by suitable spacing and tie-down. The *Nuclear Safety Guide* [9] serves as a guide for determination of safe arrangement and environmental control.

DESIGN OF CASKS

The most generally used type of cask in the US is a steel-jacketed, lead-filled right circular cylinder which encloses a similarly shaped fuel cavity, accessible by removing a shielding plug or lid. This type of cask is shipped either in a vertical position attached to a skid or in a horizontal position within a cradle. Other shapes are also used, and in some cases the cask has several small cavities.

Various combinations of the following design features have been used to meet the requirements previously discussed. Other methods may be equally acceptable.

Normal conditions

Normally, the cask dimensions are such as to preclude the possibility of appreciable interaction with other containers.

Many details of cask construction and operation not discussed here are also pertinent to safety of shipment, and are pointed out in Refs. [1] and [2].

Special features for radiation protection include (*a*) access plugs with offsets to prevent direct beaming through joints; (*b*) penetrations from the cavity to the exterior of such configuration as to prevent direct beaming; (*c*) termination of coolant drain lines at points above the cavity to prevent accumulation of radioactive sediment in unshielded locations; and (*d*) recessing of external connections and valves behind plugs of shielding material. Careful attention to points of potential beaming is required in design and fabrication.

Heat removal is accomplished by (*a*) water or other liquid that transfers heat directly from the fuel to the inner walls of the cask by either natural or forced convection; (*b*) a secondary coolant, not in direct contact with the fuel, which transfers heat from the primary coolant to external air coolers either by natural or forced circulation; (*c*) fuel baskets within the cavity in direct contact with the cavity wall; (*d*) lead that is metallurgically bonded to the shell; (*e*) fins welded or brazed inside the shell, as an alternative to bonding; and (*f*) fins welded to the outer shell.

Containment of radioactive material is achieved by (*a*) encasing failed fuel in sealed canisters; (*b*) double sealing of the cavity by a gasketed closure directly above the cavity, in addition to a gasketed shielding plug; (*c*) two gaskets on the shielding plug, with the space between the gaskets pressure-tested before shipment to make sure that neither gasket leaks; (*d*) filters on vents or pressure relief valves; (*e*) a strainer installed in the coolant outlet from the cavity.

Nuclear safety is assured by (*a*) restricting the quantity of fissile material; (*b*) holding the fuel rigidly in a safe configuration; (*c*) incorporating nuclear poisons in fuel storage baskets or parts thereof; (*d*) shipping poison rods with the fuel.

Protection against the normal rigors of transport is provided by (*a*) locking devices such as lock nuts, lock washers, and set screws to prevent loosening of parts from vibration; (*b*) tying the cask down by rods, cables, or direct attachment to the vehicle; (*c*) providing pads that cushion against vibration; (*d*) using heavy-duty draft gear for the car, and shock absorbing bumpers for the cask.

Accident conditions

Resistance to fire is accomplished by (*a*) providing spaces within the steel jacket of the cask to accommodate lead expansion; (*b*) designing the outer shell to deform sufficiently without rupture; (*c*) surrounding the cask with insulation, with suitable provision for removal of decay heat in normal cask operation.

Resistance to impact is accomplished by the following means: (*a*) the cask itself is endowed with sufficient strength and energy absorption characteristics to withstand the specified impact; (*b*) the cask is provided with a crash frame of wood or structural steel, designed to deform sufficiently to absorb a significant portion of the impact energy (the crash frame may be attached to the cask or to the vehicle); (*c*) fins, cover plates, baffles, and other devices are used partly or wholly for protective purposes; (*d*) to prevent penetration of the wall by lifting trunnions, the trunnions are either suitably positioned, protected by heavy covers, or are avoided by the use of sockets rather than trunnions on the cask.

Preservation of heat dissipation is assured by using (*a*) fuel holders that expand with increased temperature to provide good contact with inner wall of cask; (*b*) standby or duplicate mechanical devices.

Avoidance of criticality is assured by (*a*) cask construction that prevents hazardous rearrangement of fuel and poisons from impact or fire; (*b*) a fuel and poison arrangement which is safe with optimum moderation and reflection of neutrons in water; (*c*) a margin of safety that is adequate even if the shipment were commingled with other similar shipments in water.

METHODS OF ASSESSING COMPLIANCE

The criteria for licensing US cask shipments are set forth in 10 CFR Part 72 [1]. Methods used by the

USAEC for assessing compliance with certain of these criteria requiring test or analysis are as follows:

In 72.33a (beam strength) the cask regarded as a simple beam supported at its ends must be capable of supporting without rupture a uniformly distributed load of ten times the weight of the fully loaded cask. For lead-shielded casks, model testing is considered satisfactory. Alternatively, structural analysis showing that the steel shell will not buckle, assuming no strength for the lead, is acceptable.

Since the distribution of forces resulting from impact is exceedingly complex, and the strength of materials is affected by rate of deformation, an accurate analytical solution is impracticable except with certain shock-absorbing systems (72.32b, impact from drop). From many drop tests with models and casks ranging in weight from about 30 to 30 000 lb, it has been concluded that model-testing is an acceptable method of assessing gross damage [6*b*, 13–16]. With small models such as $^1/_{10}$ scale, careful attention to detail is required; thus in some instances larger models are preferable and less costly. Features not essential to cask strength may be omitted from the models. Buckling patterns for shells and structural members are sensitive to slight variations of construction and testing; however, this has appeared unimportant with regard to rupture of the shell.

Data on the stress-strain characteristics of some energy-absorbing materials at high rates of strain permit reliable calculation of impact deceleration of casks protected by such materials. Drop tests of models have shown good agreement between predicted and measured decelerations, using rubber or aluminum honeycomb as the energy-absorbing medium [6*c*]. With small casks, it has been found that crash shields of practical size and of this type can limit cask deceleration in a 30-mph impact to the order of 10 to 100*g*.

As noted in 72.34a, a shell thickness not less than that tabulated therein for the particular cask weight is considered acceptable for puncture resistance.

For fire conditions, analytical techniques are usually used to estimate cask temperature and stress distribution. Such estimates take into account expansion of lead and steel, pressure of liquid lead, and transient temperature conditions during and after the fire. Rupture of the cask shell or gross loss of shielding is unacceptable. If fuel would fail or melt at the calculated temperature, assurance must be provided that large release of radioactive materials from the cask or the formation of a critical configuration would not occur. A suitable heat insulating shield around the cask is an acceptable method of preventing excessive temperatures.

Suitable protection of an external appurtenance essential to safe operation of the cask is considered to be provided by an enclosure of steel plate not less than ½ in thick and capable of preventing damage to the appurtenance from an external force equal to the weight of the cask and applied in any direction (protection of fittings, 72.34j).

The calculative technique used to determine k_{eff} should be verified by application of the method to a system of known reactivity which is as nearly like assumed cask conditions as is feasible to produce in an experiment. Differences between test conditions and assumed cask conditions should be accounted for in a conservative manner, to ensure that k_{eff} would not exceed 0.9. The possible adverse effect of impact or fire on fuel geometry and effectiveness of poisons should be considered (nuclear safety, 72.37a).

CURRENT PROGRAM

Impact

Although models have been found satisfactory for prediction of gross damage, there are several particulars for which further investigation would be of value for cask design and evaluation. The program now in progress includes:

(*a*) Impact tests with models on flat surfaces, beams, and penetrators to determine the minimum drop height required to cause failure. Both lead-filled and solid steel casks, with and without simple crash frames, are of interest;

(*b*) Tests to determine the relationship between static load resistance and impact resistance;

(*c*) Tests to assess the relative merits of different structural details and materials of construction. The design of closures is of special concern;

(*d*) Theoretical and experimental analysis of the energy absorption of vehicles in collision, for different cask tie-down arrangements;

(*e*) Study of the requirements of the tie-down system for normal transportation, and of the value of strong fastenings for reducing the velocity of a cask before it suffers impact in a collision accident.

Fire

It is assumed that serious fires involving large casks would also usually involve liquid hydrocarbon fuels. The characteristics of such fires with JP-4 fuel have been investigated [17]; other fuels have been found to burn at about the same rate and with about the same flame temperatures [18]. Values of cask absorptivity and emissivity, the effect of soot deposits, and the magnitude of convection currents in molten lead are not known with desired accuracy for calculation of cask temperatures. Investigation now under way includes the testing of a 15-ton lead-shielded cask and several small objects in a large fire of JP-4 fuel, with

suitable instrumentation; and the development of analog computer techniques and programs for calculation of cask temperatures, with casks of different size, shape, and materials of construction.

Accident analysis

The justified degree of safety in design depends on accident frequency and severity. Many detailed studies of the frequency of accidents of different types have been made, and among those most directly applicable to the shipment of radioactive materials are Refs. [4, 5, 19]. However, little is known about the environment to which cargo is subjected in such accidents: the frequency per vehicle mile vs. speed, or frequency distribution vs. severity. Most impacts and fires are minor; the uncommonly severe accidents are those which require further study from the standpoint of damage to casks. It is noteworthy that the accident frequency for passenger trains is less than 1% of that for freight trains [5]. Identification and evaluation of factors responsible for this large difference may be fruitful.

Good administrative procedures can undoubtedly reduce the probability and consequences of accidents. Routing of shipments to avoid areas of dense population or heavy traffic, and timing of certain movements would serve both purposes. Use of trained personnel to direct damage control activities or to advise others would normally serve to reduce radiation injury and damage in accidents involving actual or potential escape of radioactive material.

Studies of various aspects of this problem are under way or under consideration, and are expected to include experimental accidents.

CONCLUSION

During the past few years much progress has been made in establishing a uniform degree of safety for casks, in developing practical methods of meeting specified requirements, and in developing data and procedures for analyzing the performance of casks. Further engineering studies are in progress. Much additional information is needed regarding the nature and frequency of accidents and regarding the consequences of cask failure before the various risks relating to over-all long-range economy can be evaluated. As cask shipments become more numerous, experience should accumulate to permit more complete analysis of design criteria and to indicate areas of design improvement.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/279 Etats-Unis d'Amérique

Problèmes de sécurité posés par le transport du combustible irradié

par J. W. Langhaar *et al.*

Le transport du combustible irradié, comme celui d'autres marchandises jugées dangereuses, met en jeu des considérations de sécurité et d'économie. Il importe au plus haut point d'assurer la protection de la population contre les risques radiologiques pour le cas où un accident se produirait. Cependant, les renseignements que l'on possède sur la probabilité et les conséquences des accidents sont si fragmentaires qu'à l'heure actuelle il appartient aux pouvoirs publics et aux techniciens d'apprécier quel degré de sûreté est justifiable. Le fait que les risques sont mal connus de la population et que, contre certains d'entre eux, l'individu n'est pas prévenu par des signes physiologiques, fait nettement pencher la balance du côté de la sécurité.

Aux Etats-Unis, dix accidents dus à des défaillances de châteaux de transport au départ, en transit ou à l'arrivée ont été signalés au cours de la période 1957-1962. Dans six cas, il n'y a pas eu fuites de produits radioactifs; dans les quatre autres, une certaine quantité d'eau contaminée s'est échappée, mais il n'y a pas eu de radiolésions. Heureusement, dans aucun de ces accidents un château n'a été exposé à un choc brutal ou au feu.

L'ensemble des considérations dont il faut tenir compte dans l'étude des châteaux de transport a un caractère assez exceptionnel; doivent être assurés la protection contre les rayonnements, l'évacuation de la chaleur engendrée par les produits de fission, la prévention des fuites de produits radioactifs en transit, l'isolement du contenu en cas d'accident grave et, en certains cas, la prévention de la formation d'un état critique. Les auteurs présentent diverses méthodes qui ont été appliquées ou sont à l'étude.

Les méthodes utilisées pour calculer la résistance au choc des châteaux non protégés ne sont pas sûres; toutefois, les essais de résistance pratiqués sur des maquettes et des châteaux en vraie grandeur ont donné des résultats concordants, les échantillons utilisés allant de 13,6 kg (30 lb) à 13 600 kg (30 000 lb). Les auteurs ont procédé à une analyse théorique de l'utilisation de matériaux qui absorbent les chocs pour la protection des châteaux et ont vérifié expérimentalement les conclusions auxquelles ils étaient parvenus.

Ils décrivent les travaux entrepris afin d'obtenir des données pour le calcul des dommages causés aux châteaux de transport par le feu. Si l'on constate qu'il se révèle difficile ou onéreux de les rendre très résistants au feu, il faudra étudier attentivement la fréquence et les conséquences probables d'une défaillance d'un château. Les auteurs présentent certaines statistiques d'accidents.

La façon dont les châteaux sont montés sur les véhicules peut avoir beaucoup d'importance en cas d'accident. Cette question est à l'étude. Les auteurs indiquent les principales considérations qui s'y rapportent.

Certains principes applicables à la construction des châteaux de transport ont été énoncés par l'Interstate Commerce Commission des Etats-Unis et le Bureau of Explosives en tient compte lors de l'octroi des permis d'utilisation. Des prescriptions plus détaillées figurent dans le projet de règlement fédéral, 10CFR, Partie 72, que la Commission de l'énergie atomique des Etats-Unis étudie actuellement en vue d'une revision éventuelle. L'American Standards Association a publié un manuel intitulé « Recommandations pratiques générales concernant l'étude et l'utilisation de conteneurs destinés à l'expédition et au transport de combustible solide irradié provenant de réacteurs nucléaires ».

Les auteurs résument les conditions et recommandations importantes que contiennent ces documents.

Différentes mesures d'ordre administratif peuvent contribuer à assurer la sûreté du transport. On peut citer le choix de l'itinéraire, du type de véhicule, de l'heure de certains déplacements, de la place du wagon dans un train, ainsi que les limitations de vitesse et parfois le convoiement. Toutefois, il faut de manière générale s'efforcer de fabriquer du matériel qui puisse être manutentionné sans danger à l'aide des méthodes courantes.

A/279 США

Вопросы безопасности при транспортировке облученного ядерного топлива

Дж. У. Лангхар *et al.*

Транспортировка облученного ядерного топлива, как и других опасных веществ, должна производиться с учетом требований безопасности и экономической стороны вопроса. Особенно важно защитить население от радиационного воздействия при аварии во время транспортировки. Однако имеющихся данных о вероятно-

сти и последствиях таких аварий настолько недостаточно, что в настоящее время оценка достаточности принимаемых предохранительных мер предоставляется административному и инженерному персоналу. Неосведомленность большинства людей об опасностях, связанных с транспортировкой облученного топлива, и отсутствие предупреждающих физиологических признаков таких опасностей намного усложняют задачу обеспечения защиты и подчеркивают необходимость тщательной проработки вопросов безопасности.

С 1957 по 1962 год в Соединенных Штатах имели место 10 официально зарегистрированных аварий или неисправностей контейнеров, содержащих радиоактивные вещества, во время их транспортировки или на станциях. В шести из этих случаев не было утечки радиоактивности, а в четырех, хотя небольшое количество воды, содержащей радиоактивные материалы, и пролилось, все обошлось без лучевых повреждений. К счастью, ни в одном из этих случаев контейнер не подвергся сильному толчку. Потери также отсутствовали.

Сочетание факторов, которое приходится принимать во внимание при проектировании контейнеров, до известной степени уникально. Требуется предусмотреть защитный экран, удаление тепла, вырабатываемого продуктами деления, предотвратить утечку радиоактивных материалов во время транспортировки, удержать их в случае серьезной аварии, а иногда и предупредить возможность образования критической массы. В докладе рассматриваются различные методы решения этих задач, часть из которых в настоящее время уже применяется, а часть находится в стадии обсуждения.

Методы расчета сопротивления непредохраненных контейнеров воздействию ударов недостаточно точны, однако результаты экспериментов с падающими моделями и контейнерами полного размера находятся в хорошем соответствии с теорией для веса последних от 13,6 до 13 600 *кг*. Был проведен теоретический анализ возможности применения амортизирующих материалов для предохранения контейнеров, результаты которого были проверены экспериментально.

В докладе описаны работы, проводимые в настоящее время с целью получения необходимых данных для определения влияния пожара на контейнер. Если будет установлено, что обеспечить достаточную огнеупорность контейнера слишком трудно или дорого, то придется тщательно рассмотреть вопрос о том, как часто будут повреждаться контейнеры и каковы будут последствия поломки контейнеров в результате пожара. Рассматриваются некоторые статистические данные о происшедших авариях.

Используемый метод крепления контейнера при транспортировке может сильно повлиять на последствия аварии. Этот вопрос в настоящее время находится в стадии изучения. В докладе рассматриваются главные стороны данного вопроса.

Некоторые соображения по расчетам безопасности предусмотрены правилами, выработанными Межштатной торговой комиссией США, и обсуждались в Бюро взрывчатых веществ. Более детальные требования изложены в постановлении 10 CFR, статья 72, которое в настоящее время рассматривается в Комиссии по атомной энергии США; Американская ассоциация стандартов выпустила в печать справочник под заглавием «Американские стандарты в области проектирования и эксплуатации транспортных контейнеров для твердого облученного топлива ядерных реакторов». Дается резюме главных требований и советов, содержащихся в этих документах.

Ряд других факторов также влияет на степень риска при перевозке; они включают выбор дороги, тип перевозочного средства, расписание передвижения, место вагона в поездке, ограничение скорости движения и, в некоторых случаях, сопроводительный конвой. Однако главное внимание должно быть обращено на то, чтобы создаваемое оборудование могло допускать безопасное с ним обращение в нормальных условиях.

A/279 Estados Unidos de América

Los aspectos de seguridad en el transporte del combustible nuclear irradiado

por J. W. Langhaar *et al.*

En el transporte del combustible irradiado, como en el caso de otras mercancías consideradas peligrosas, hay que sopesar la seguridad y la economía. La protección del público de los riesgos radiactivos, en caso de accidente en el transporte, tiene una importancia especial. Sin embargo, nuestros conocimientos respecto a la probabilidad y a las consecuencias de los accidentes son tan limitados que el actuar con un grado justificable de seguridad es, actualmente, una cuestión de buen juicio administrativo y técnico. El desconocimiento de los riesgos por parte del público y la falta de señales fisiológicas que indiquen algunos de estos riesgos inclinan fuertemente la balanza del lado de la seguridad.

En los EE.UU., del año 1957 al 1962, hubo, que se sepa, diez accidentes o averías en los recipientes de embarque ya en tránsito, ya en los terminales. En seis de ellos no hubo escape de material radiactivo; en los otros cuatro, se vertió algo de agua contaminada, pero nadie sufrió daños. Afortunadamente, en ninguno de los accidentes los recipientes de embarque se incendiaron o recibieron golpes fuertes.

La combinación de los factores que intervienen en el diseño de los recipientes de embarque no es corriente, entre ellos se encuentran: la previsión del blindaje,

la extracción del calor generado por los productos de fisión, el evitar el escape de los materiales radiactivos en tránsito, la contención de la carga en caso de accidente grave y, en algunos casos, el prevenir la formación de un conjunto crítico. Se discuten varios métodos que se han utilizado o que se considera satisfacen dichos objetivos.

No se puede confiar en los métodos de cálculo de la resistencia de los recipientes de embarque a los golpes; sin embargo, las pruebas de caída con modelos y con recipientes de tamaño real dan una buena correlación para modelos cuyo tamaño varía entre 13,6 kg (30 lb) y 13 600 kg (30 000 lb). Se ha analizado teóricamente la utilización de materiales absorbentes del choque para proteger los recipientes y se han comprobado experimentalmente las conclusiones.

Se describen los trabajos que se llevan a cabo para adquirir los datos de cálculo de los efectos del fuego sobre los recipientes de embarque. Si se encuentra que el construir recipientes a prueba de incendios es muy difícil o costoso, habrá que considerar con cuidado la frecuencia esperada y las consecuencias de las averías en los recipientes. Se presentan estadísticas de accidentes.

Los métodos para instalar los recipientes en los vehículos pueden tener repercusiones considerables sobre las consecuencias del accidente. Este tema se encuentra en estudio. Se señalan las consideraciones más importantes.

Ciertos aspectos del proyecto quedan cubiertos por las normas de la U. S. Interstate Commerce Commission y tenidas en cuenta por el Bureau of Explosives al conceder los permisos. Se establecen requisitos administrativos más detallados en las normas federales, 10CFR Part 72, propuestas que, sin embargo, está revisando la USAEC. La American Standards Association ha publicado una guía titulada « Guías recomendadas por la American Standards Association para proyectar y manejar los recipientes de embarque de combustible nuclear sólido e irradiado » (« American Standard Recommended Practice for Design and Operation of Shipping Containers for Irradiated Solid Fuel from Nuclear Reactors »). Se resumen los requisitos y recomendaciones importantes de estos documentos.

Se pueden usar varios procedimientos administrativos para ayudar a garantizar la seguridad del envío. Entre otros se encuentran la ruta, el tipo de vehículo, la programación de ciertos desplazamientos, la situación del vagón en el tren, la limitación de velocidad y, en algunos casos, la escolta por personal idóneo. Un objetivo amplio es, sin embargo, el suministro del equipo apropiado para el manejo sin riesgos, siguiendo procedimientos normalizados.

Transport of irradiated nuclear fuel on the European continent

By F. R. Marcus and J. Asyee *

The irradiated fuel resulting from the operation of reactors on the continent of Western Europe calls for the transportation and management of considerable quantities of fission products and fissile material. Regardless of whether the irradiated fuel is reprocessed or not, transport involving a potential hazard to the population must usually take place, and efforts to ensure that safety regulations are met to minimise this hazard result in complications in achieving such transportation.

The removal of irradiated fuel from the reactor site can be justified on economic or safety grounds. If combined reprocessing and transport prices permit economic re-use of fissile material, or if centralised storage of the bulk fission products is required, transport procedures will follow the pattern existing in countries where nuclear energy has already reached an advanced stage of development.

The size and location of processing plants influence the costs associated with this part of the fuel cycle. It has been shown that in a future nuclear energy system, reprocessing plants with capacities of the order of several tons per day for near-natural uranium will be required, and that reprocessing costs may be reduced by an order of magnitude from those at present known for plants having capacities of 0.3–1 ton/day. On the other hand, relatively low transport costs (such as can be achieved on the European continent) in conjunction with to-day's plutonium and ^{235}U prices justify processing in smaller plants, and even allow economic reprocessing of natural uranium fuel.

A considerable number of research reactor using highly enriched uranium fuel have been installed on the continent, and transport problems are now being encountered. At present there are no continental plants in operation for reprocessing MTR fuel, and shipment is therefore done to overseas installations. The largest research reactors on the continent and their fuel transport requirements are listed in Table 1. Processing is justified because of the high fuel "rest value", despite considerable transport costs. Considering existing and planned cask capacity, about 100 cask journeys, involving approximately 20×10^6 c, can be expected in 1965 with MTR fuel.

Only a few power reactors are at present in operation, and no fuel has been transported from them as yet. Table 2 shows important reactors expected to be in operation before 1969, and their anticipated discharge. According to this information the transportation of more than 300×10^6 c per annum will be taking place some 4 years from now.

A graph of the annual power reactor fuel transport demand is given in Fig. 1. This shows the annual quantity of irradiated fuel available for shipment, expressed as curie-activity, assuming a cooling time of 150 days.

As a result of reprocessing, transportation of decontaminated fissile material will become necessary. Although not discussed in detail here, it must be emphasized that bulk shipment of low enriched decontaminated uranium preferably should be limited to short distances and that transport costs influence the decision to re-enrich the material either by blending (done on the continent) or by diffusion (done overseas).

Transport of low active solid waste material takes place on a small scale and is not significant in the overall picture. The present trend in high level waste management is decay storage followed by solidification and final storage, possibly at centrally-located national or international sites. Although after 10 years decay storage the activity is reduced to less than 5% of that present in irradiated fuel cooled for 150 days, and in spite of the absence of criticality problems, waste transportation may be expected to contribute significantly to the overall fuel cycle costs. There is no continental experience in this field.

The first transportation of irradiated fuel from a continental reactor took place in 1958, when elements from the French EL2 reactor were sent to the United Kingdom for processing. Apart from the transportation of test elements and the limited transport of fuel from the EL1 reactor for reprocessing at Eurochemic in Mol and from the Ispra reactor to Casaccia, only the shipments from the Danish DR3 to the United Kingdom and from the Swedish R2 to the United

* Eurochemic, Mol, Belgium.

One of the Joint Undertakings of the European Nuclear Energy Agency.

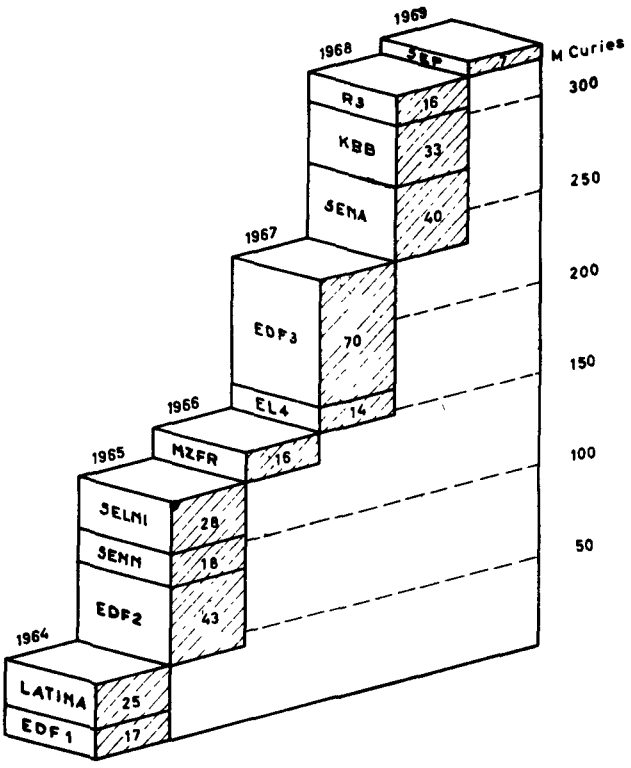


Figure 1. Scheduled total activity involved in the transport of irradiated fuel from continental power reactors

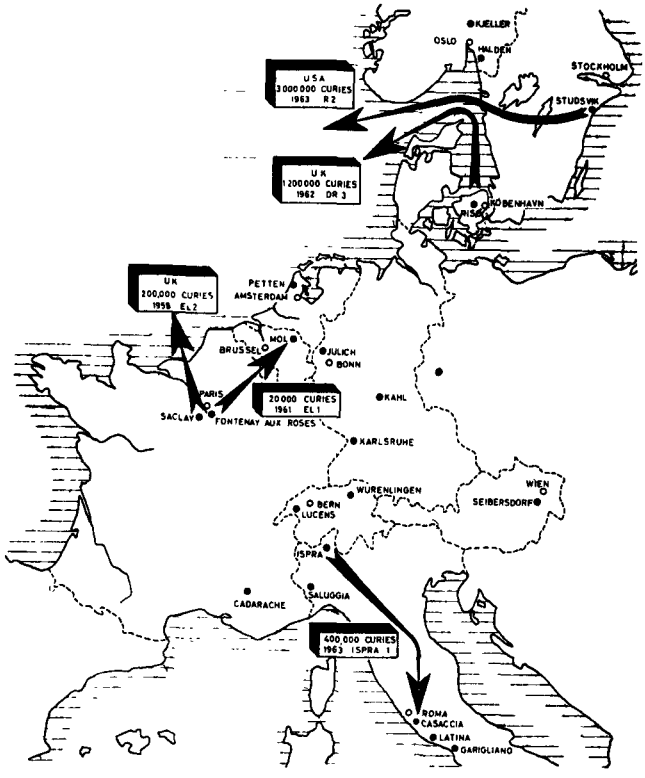


Figure 2. Shipments of irradiated fuel from continental reactors up to April 1964

States were on a large scale. Some details of these transport operations are collected in Table 3 and Fig. 2. No incidents have been reported.

The experience from this Italian shipment is inconclusive, as efficient escorts were requested, necessitated by the road regulations at the time of shipment. Danish and Swedish experience is interesting in that all transport arrangements were made by the national atomic energy authorities, and costs were similar to those of conventional transport. Italian shipments were insured at a premium equivalent to the basic transport cost, while the Danish and Swedish Governments covered the risks whilst the casks remained within their frontiers. The DR3 casks were approved by the UK and the R2 casks by competent US authorities. No difficulties were met in obtaining national permits. Specially designed casks, approved in France, were used for transportation of the EL1 elements. These shipments were escorted only within the frontiers of Belgium.

Transportation of irradiated fuel elements from research reactors within the EURATOM countries is to be carried out in the near future, and appreciable amounts, especially from the reactors HFR and BR2, are available for transport. A collection centre is planned for a number of German research reactors to which individual reactors can send their elements for interim storage, and where inactive ends can be removed.

Table 1. Anticipated transport schedule for large research reactors (above 10 MW) with regular fuel discharge

Country	Reactor	Power (MWth)	^{235}U (g per element)	Estimated annual quantities		
				Elements	kg ^{235}U	Cask journeys
Belgium	BR2	50	250	160	27	12
Denmark	DR3	10	115	70	6	3
France	SILOÉ	10	196	100	14	17
	PÉGASE	30	380	110	30	6
Germany	DIDO (Jülich)	10	115	75	7	4
Netherlands	HFR	20	125	130	11	8
Sweden	R2	30	130	320	30	36

Table 2. Important power reactors with regular fuel discharges

Country and reactor	Power (MWth)	Estimated first discharge	Estimated annual quantities			
			ton U	kg ^{235}U	kg Pu	10^6 curies
Belgium						
SENA	950	1968	16	330	68	40
France						
EL4	250	1967	7	40	26	14
EDF1	300	1964	30	135	60	17
EDF2	800	1965	77	345	154	43
EDF3	1 480	1967	120	540	240	70
Germany						
VAK	60	1964	2	20	5	2
KRB	801	1968	14	128	57	33
MZFR	200	1966	13	38	31	16
Italy						
SENN	506	1965	12	108	54	18
SIMEA	705	1964	70	294	132	25
SELNI	825	1965	13	261	77	28
Netherlands						
SEP	175	1969	4	44	14	7
Sweden						
R3	65	1968	3	15	7	16
Switzerland						
Lucens	30	1967	3	17	6	2
Total	6 900		385	2 300	940	330

Irradiated fuel from the French EL3 reactor and the Ispra 1 is also expected to be shipped soon.

The first shipments from power reactors will involve BR3, VAK and EDF1 fuel; but the bulk shipments to take place in the coming years will involve fuels from such reactors as SIMEA, SENN, EDF2, SELNI, MZFR, EL4, KRB and SENA (Fig. 1).

REGULATIONS

In line with regulations governing the transport of other dangerous materials, a system of national and international rules, enforced by competent authorities, is being developed for the transportation of radioactive material.

Regulations for the Safe Transport of Radioactive Materials were issued by the IAEA in 1961 ("Vienna regulations") and were recommended for use as a basis for national and international regulations. Some general standards for the design of casks for irradiated fuel were included in the Agency's regulations. However, only little practical advice for cask design, testing and consequently approval was given. Therefore, other rules — especially the USAEC and the UKAEA design practice — were adopted in some countries.

A revised edition of the Vienna regulations is being prepared. Whereas in the 1961 edition, irradiated fuel casks were required to withstand the "maximum credible accident", in the revised edition that difficult concept will be replaced by specified test requirements which should give an adequate basis for design and approval.

The national atomic energy laws, listed in Table 4, do not generally give details concerning the transport of irradiated fuel, and special approval by competent authorities (see Table 5) is required in each case; but modalities similar to those contained in the Vienna regulations are already applied in practice.

International transport organizations also mainly follow the Vienna recommendations. Thus class IVb of RID (Règlement international concernant le transport des marchandises dangereuses), covering rail transport, requires the approval of casks in the country of origin only, while each country involved may impose special modalities for the transport. The ADR (Accord européen relatif au transport international des marchandises dangereuses par route) and ADN (Accord européen relatif au transport international des marchandises dangereuses par voie navigable) tend to follow the same trend; but at present, international road and water transports may still require a great number of individual permits to be given, e.g. by local authorities. This situation at present favours rail transport. The IATA regulations include provisions derived from the Vienna regulations, but air transportation is not considered practicable for irradiated fuel.

Table 3. Shipments of irradiated fuel from continental reactors up to April 1964

Reactor	EL2	EL1	DR3	ISPRA-1	R2
First transport	1958	1961	1962	1963	1963
From	Saclay	Fontenay	Risø	Ispra	Studsvik
To	UK	Mol	UK	Casaccia	USA
Mode of transport	road, ship	road	road, ship	road	road, rail, ship
Approx. distance on continent (km)	—	400	40	825	400
Number of transports	3–4	2	6	3	5
Fuel weight per transport (kg)	—	1 000	110 (U + Al)	45 (U + Al)	160 (U + Al)
Activity per transport, curies	—	< 10 000	200 000	120 000	600 000

Table 4. Legal situation in April 1964 in some European countries

Country	Nuclear law, or legislation on nuclear installations	Law regarding 3rd, party liability	Insurance coverage required	Nuclear pool
Belgium	[a]	For CEN, 16.5.62	10×10^6 CEN	Syndicat belge d'Assurances nucléaires, Bruxelles
Denmark	16.5.1962	Act. of 16.5.62 [e]	10 M $	Dansk Atomforsikrings Pool, København
France	11.12.63 [b]	—	[g]	Pool français d'Assurances des risques atomiques, Paris
Germany	23.12.59 [c]	Part of nuclear law [f]	[h]	Deutsche Kernreaktor-Versicherungsgemeinschaft, Köln
Italy	31.12.62	Part of nuclear law	5 M $	Pool Italiano per l'Assicurazione dei Rischi atomici, Roma
Netherlands	Draft law	Draft	15 M $	Nederlandse Pool voor Verzekering van Atoomrisico's, Amsterdam
Sweden	1.6.56 [d]	Act. of 3.6.60 [e]	5 M $	Svenska Atomförsäkringspoolen, Stockholm
Switzerland	23.12.59	As nuclear law	10 M $	Schweizer Pool für die Versicherung von Atomrisken, Zürich

[a] Arrêté royal of 28.2.63 contains administrative regulations concerning radioactive transports;

[b] Transport rules are prescribed in decrees in *Journal officiel*; for land transport see in particular *Journ. off.* of 29.8.1963.

[c] Irradiated fuel shipments must conform to USA federal register part 72.

[d] Special transport rules, issued 1.6.1962.

[e] At present the state is self-insuring.

[f] Carrier must take out insurance, following Financial Security Ordonance of 22.2.1962.

[g] Routinely between 5 and 15 M $.

[h] 0.25 M $ for 1 kg ^{235}U, 0.025 M $ for each further kg.

Although not required by most of the national and international rules, escorts have been requested for some early shipments. Escorts reduce the probability of accidents and ensure that prompt measures are taken if abnormal situations do occur. The tendency is to rely on the inherent safety of cask design; thus, in general, no escorting of future consignments is envisaged.

While property insurance of irradiated fuel follows conventional lines, some problems exist in obtaining coverage at a reasonable premium against the consequences of a nuclear incident during transport. The 1960 OECD (Paris) Convention and its 1964 Additional Protocol, as well as the 1963 Convention Supplementary to the Paris Convention establishing supplementary coverage by all States up to $120 million, have not yet entered into force, nor has the 1963 Vienna Convention. The Paris and Vienna Conventions define the liability (imposed on the operator of the nuclear installation), the amount he must cover ($5 to 15 million) and the duration (10 years after the incident). At present, each country has its own regulations (Table 4); certain States, as operators of the reactors, are self insuring, whilst other States cover risks above the limits insured by the person liable during the transport.

Transport insurance is presently available from "Nuclear Pools", which were created in 1957 by insurance companies in individual countries in order to distribute the risk. Nuclear Pools are also listed in Table 5.

COSTS

At present, only transport by road and rail are considered for shipments of active fuel on the continent. As only a few reactor sites are directly accessible by rail, transport entirely by this means is unlikely and "mixed" transportation requires the establishment of transfer points.

There has been too little experience to quote exact all-inclusive transport costs. However a number of components are available, and reasonable estimates can be made on the basis of present experience, and firm offers obtained. The main items include the basic transport costs, fees for approval and special arrangements, cask and possibly escort costs, and insurance premiums.

Transport firms competent to handle nuclear materials exist in several countries. Three such firms were selected — from France, Germany and Italy — for transport of fuel from EURATOM research reactors following invitation to tender. Experienced firms from overseas also offer their services for cask hire and transport. Finally, some reactor owners have arranged transportation themselves.

To facilitate economic studies it would be convenient to express transport costs in $/kg of fuel transported, or mills per kWh. As the fuel itself constitutes only a fraction of the total weight to be transported, and as the total weight essentially determines the transport costs, the cask/fuel weight ratio, i.e. the pay load, thus becomes important.

Table 5. Competent authorities controlling the transport of irradiated fuel in a number of countries

Country and competent authority	Advisory body
Austria. — Bundesministerium für Verkehr und Elektrizitätswirtschaft, Gauermanngasse 2–4, Vienna	—
Belgium. — Ministère de la Santé Publique et de la Famille; Institut d'Hygiène et d'Epidémiologie, 14, rue Juliette-Wytsman, Brussels	CEN
Denmark. — Sundhedsstyrelsen, St. Kongensgade 1, Copenhagen	—
France. — Ministère des travaux publics et des transports, Transport des matières dangereuses, 244, bd. Saint-Germain, Paris (VII[e])	CEA
Germany. — Bundesminister für Verkehr [a], Sternstrasse 100, Bonn	PTB (a) BAM
Italy. — Ministero del'Industria e del Commercio [b], Via Molise 6, Rome	CNEN
Netherlands. — Ministerie van Verkeer en Waterstaat, Binnenhof 20, The Hague [c]	—
Norway. — Statens Radiologiske-Fysiske Laboratorium, Ullernshausseen 70, Oslo	IFA
Portugal. — Direction Générale des Transports Terrestres, Rue de S. Mamede, ao Caldas 21, Lisbon	
Spain. — Direccion General de Transportes Terrestres, Ministerio de Obras Públicas, Madrid	—
Sweden. — Kung. Medicinalstyrelsens Strälskyddsnämnd Radiofysiska Institutionen, Karolinska Sjukhuset, Stockholm	AB Atomenergi
Switzerland. — Der Delegierte für Fragen der Atomenergie, Effingerstrasse 55, Bern	—

[a] In certain cases: Physikalisch-Technische Bundesanstalt, Bundesallee 100, Braunschweig.

[b] CNEN for RID.

[c] For RID only, otherwise also Ministerie van Sociale Zaken.

Transport and cask costs expressed as $/kg of fuel vary considerably for MTR fuel in comparison with power reactor fuel. As shielding thickness directly affects the cask weight, a comparison on a basis of $/curie transported might, in fact, be preferable.

For transport by rail, the basic cost is of the order of $0.03–$0.05 km/ton cask. However, the cost of a special train may be about 0.1 $/km-ton cask. The cost of handling and transferring a cask to and from the reactor and to the rail-road transfer point has been quoted as 5–20 $/ton cask.

For transport by road, the basic cost is of the order of $0.02–$0.03 km/ton cask. However, there is an additional daily tariff for waiting time. If a special trailer is exclusively used for a specific shipment, high initial cost and limited utilization rapidly leads to an increase in the basic cost, which has in such cases been quoted as 0.15 to 0.2 $/km-ton cask.

For transport by road, basic costs are independent of cask weight within certain limits (i.e. costs may be the same for, say, a 10 t and a 15 t cask of the same vehicle is used).

Fees for approval and special arrangements are not yet available, but they are not expected to contribute significantly to the overall cost.

The cost of escorts by road and rail have been quoted at 0.3–1 $/km, depending on the type of escort and the average speed. In one instance where transportation speed was slow and an excess of escorts were provided, the cost was 4 $/km.

Insurance premiums at present vary mainly according to distance or duration, to whether the transport is escorted, and to the total number of curies (even though casks are able to withstand a "maximum forseeable accident"). An approximate idea of the cost of premiums can be calculated using the formula: $a + bD + cA$ ($), where D = distance in km and A = activity in 10^5 curies, and assuming $a = 600$, $b = 0.3$ and $c = 7$. This gives premiums of the order of $650–$1 500 per shipment. However, methods of calculation have neither been published nor definitely fixed. Although such premiums do not prevent the economic transportation of fuel, they are considered to be too high; but recently, premiums appear to show a downward trend.

Price quotations for transport service from specialized firms include the above-mentioned items and generally depend on the frequency of the shipments which reflects the depreciation of special material, especially where the service includes hire of the cask and the use of special trailers.

Cask cost per kg of uranium shipped mainly depends on the basic cask price, cask life time, utilization and, naturally on the pay load.

In the past, casks have been manufactured in Europe at a cost of about 1.5 $/kg. Design cost quotations for cask design and licencing range from $5 000–50 000 per design, where the degree of standardization, available "know-how", and even regulatory requirements affect the cost. While cask life-time cannot be stated to-day, it seems realistic to assume amortization over approximately 10 years and to add 25% of the cask cost for maintenance. Thus a yearly sum of 18% of the purchase cost must be included in the calculations.

The cask pay load depends mainly on the fuel activity, geometry and heat transfer. For power reactor fuel, cask/fuel ratios of between 85 and 15 * are reported; for MTR fuel, cask/alloy ratios vary between 300 and 75 **. High utilization factors are realized when casks are used to serve large reactors which continuously require one or more casks, or if casks are designed to serve more than one reactor.

* Kg loaded cask per kg U transported.

** Kg loaded cask per kg (U + Al) transported.

As a rule-of-thumb guide, a cask for low enriched fuel may be considered adequately utilized if the number of journeys made during its lifetime exceeds twice the cask/fuel ratio.

Overall transport costs of 2–16 $/kg of power reactor fuel and 25–400 $/kg MTR alloy are assessed from the above data, providing that only continental transportation is considered.

CASKS

A number of casks for transport of irradiated fuel from continental reactors exist or will become available soon. Table 6 shows the total weight and other characteristics of such casks. The majority have been designed by European firms, and examples of such designs are given in [1].

Cask design must conform to the regulatory requirements, in particular for decay-heat removal and mechanical resistance, as well as criticality and radiation safety.

Internal heat removal should be adequate, even when cooling liquids are lost as a result of an accident, in order to avoid excessive temperature rise of the fuel elements leading to a possible failure of the can. The necessity of this requirement is still being discussed by safety experts. Because of the spacing thus necessary between fuel elements the cask/fuel ratio is strongly affected. The use of tight fitting baskets and inserts to improve heat transfer between element and basket or basket and cask, and the application of special materials to improve heat transfer between components are now being considered to achieve optimum cask design.

External heat removal has to be such as to avoid high external temperature under adverse transport conditions. With metal cooling fins, casks can thus dissipate about 1 kW per m^2 of cask surface. For each type of fuel, a maximum cask size can therefore be defined that just permits removal of the decay heat, even with close spacing of the fuel. If the cask size is increased, the larger cavity can only be utilized to a limited extent because of the heat dissipation problem.

Proof of mechanical resistance is frequently demonstrated by prototype testing. For example, shielding and containment efficiency should not be impaired after a free drop on to a defined target and subsequent exposure to fire. Heavy steel lining or cast iron is being used to meet these demands. Further protection is obtained by the use of metal or wooden shields, which act as shock absorbers and as heat barrier in case of fire. "Tie-down" systems or constructional features of the vehicles have been taken into consideration as part of the required mechanical structure — this however calls for special precautions during loading and unloading operations.

Table 6. Casks for transport of continental reactor fuel

(Information available up to April 1964)

No.	Reactor	Number of casks	Cask weight (ton)	Elements per cask	Cask/fuel ratio	Overall size L/D (m)	Approved by [e]	Status [f]
1	BR2	1	17.4	13	270	2.3/1.2	USA-GB	O
2	BR3-VAK	1	26	6	69	3.0/1.4	USA-GB	S [e]
3	DR3	1	14.5	25	130	1.4/1.2	GB	U
4	EL1	3	2.1	12	7	2.3/0.37	F	U
5	EL2	6	7	100	20	0.9/0.9	F	U
				328	7			
6	EL3	1	7.5	60	84	1.0/0.9	F	U
				84	60			
7	EDF1	3	22	128	22	2.1/1.6	F	O
8	EDF2	—	55	216	25	2.4/2.2	F	O
9	MÉLUSINE	[a]	9.7	9	210	1.5/0.9		[g]
10	PÉGASE	[b]	18	32	72	1.7/1.5	F	S
11	FRM	2	12	9	330	1.7/1.1		O
	FRH	1	14	26	135	1.2/1.4		S
12	[d]	1	17.7	1	—	3.3/0.8		S
13	ISPRA-1	10	10.5	12	175	1.4/1.0	US [h]	U
14	SIMEA	3	45	175	25	2.6/2.2	US	Built
15	HFR	[c]	17.4	16	270	1.8/1.3	USA-GB	S
16	R2	10	11.5	9	285	1.4/1.0	USA-S	U
17	R3	1	28.8	7	37	5.4/1.0	S	O
18	R3	1	15.5	1	140	5.8/0.5	S	O

[a] Several; also used for ISPRA-1, HFR, SILOÉ, TRITON in slightly modified form.

[b] Also for SILOÉ.

[c] Other types considered.

[d] Universal one element cask.

[e] International traffic code.

[f] O = ordered; S = design stage; U = in use.

[g] Some in use.

[h] Three casks only.

Optimum cask size not only depends on the decay heat removal and fuel geometry, but also on such factors as available crane capacity, conventional traffic regulations and utilization of the cask. In [2] an example is given of the specific optimization study considering such aspects.

The use of common casks for several reactors with identical or similar fuel elements leads to improved utilization, which in turn could justify the slight deviations from optimum design which may become necessary. This precept has been recognized for a number of recently-developed casks (viz 2, 9, 10, 12, 13 in Table 6).

A further degree of standardization seems feasible and economically justifiable. A study of the continental power reactor fuels requiring shipment up to 1969, and for which no casks have yet been foreseen, indicates that transport to a centrally-located destination can be provided using two types of casks only. Starting with one of each type to transport 23 tons of uranium in 1966, a gradual increase in the stock of casks to three type 1 casks (of about 40 tons) and two type 2 casks (of about 20 tons), excluding eventual spare capacity, would enable the transportation to be achieved in 1969 of almost all of the 80 tons uranium for which no casks at present are foreseen.

CONCLUDING REMARKS

The technical feasibility of the transport of irradiated fuel has been discussed above. Its economic significance is related to the contribution of reprocessing costs in the fuel cycle. Proposed all inclusive European reprocessing prices for fuel from water cooled power reactors vary, according to burn-up and quantity, between 30 and 60 $/kg U and thus consume 15–30 % of the fuel cycle costs assuming a plutonium credit of $10 per gram.

Considerable reduction of the costs of reprocessing is expected to result from increased plant capacities. Figures as low as 5 $/kg U for a 10 ton daily throughput have been quoted; then these may become lower than the cost of fuel transportation.

Transportation costs are expected to be 2 $/kg U for short distances and 10 $/kg U for a distance of 1 000 km, provided that casks and trailers are properly utilized, and provided that the initial period in which special arrangements will be required has been passed. These figures represent from 1 to 16-5 % of the fuel cycle cost.

It thus becomes evident that the transportation costs of the irradiated fuel could, when shipped over larger distances on the continent, offset the lower reprocessing costs obtained from centralizing large reprocessing capacity, i.e., an economical optimum capacity for a reprocessing plant must exist in respect to the total nuclear power installed in a certain area.

In Fig. 3, the total of reprocessing costs and average transport costs has been plotted as a function of the total installed nuclear power for (*a*) one centrally-located reprocessing plant and an average distance of 1 000 km to the reactors on the area under considera-

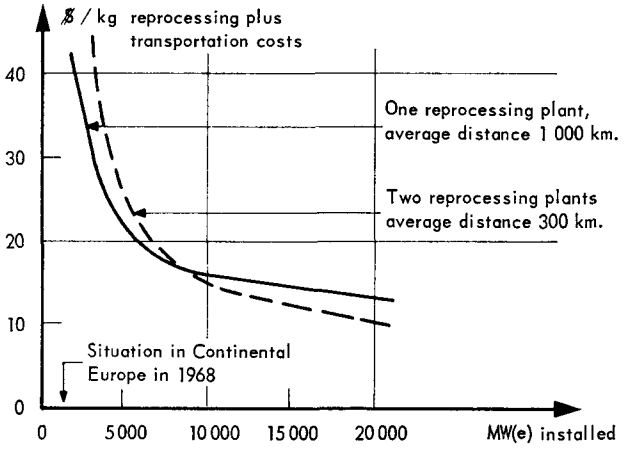


Figure 3. Reprocessing plus transportation costs as a function of installed nuclear power

tion, and (*b*) two reprocessing plants of equal capacity, the average distance to the reactors being reduced to 300 km.

Case (*a*) could very well represent the situation in continental Europe. Considering that not more than 1 500 MW(e) installed nuclear power would require a reprocessing service in 1968, it would seem that, for the time being, no economic advantage would be gained by decentralizing reprocessing plants. Moreover, if the present rate at which nuclear power is expanding in Europe increases by even a factor of three during the next two decades, the construction of more than one large processing plant would not be justified within the next fifteen years.

It is apparent that there is a strong incentive to keep overall transportation costs down in view of future fuel cycle costs. This may be achieved not only by optimum cask design and improved cask utilization, but by avoiding superfluous requirements in the drawing up of regulations. In the next few years, development on the continent is expected such that transport of irradiated fuel changes from its present embryonic state, where special arrangements and prices are applied, to a routine system with the minimum of extra requirements (because of the inherent safety laid down in the regulations) and at costs approaching those for conventional shipments.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/422 Danemark

Transport de combustibles nucléaires irradiés sur le continent européen

par F. Marcus et J. Asyee

Le mémoire donne un aperçu général de l'état actuel du transport en Europe des combustibles irradiés. Ses aspects technique, administratif et économique, et son incidence sur le coût du cycle de combustible, sont traités. Dans quelques années, le transport de combustibles irradiés, de matériaux fissiles purifiés et éventuellement de déchets actifs, entre les différents pays du continent, deviendra habituel.

Depuis 1963, les éléments de combustible MTR ont été réexpédiés systématiquement hors du continent, pour retraitement. Les expéditions importantes des premiers réacteurs de puissance n'ont pas encore commencé, mais il est prévu que, avant 1968, 270 tonnes de combustibles irradiés seront déchargées des réacteurs de puissance continentaux. La valeur de la plupart de ces combustibles justifie leur retraitement du point de vue économique; on prévoit donc un accroissement du volume des combustibles à transporter.

On donne un aperçu des législations nationales et des règlements de transport ainsi que des procédures d'autorisation nécessaires, établies dans un certain nombre de pays européens. Les règlements internationaux et les principes de sécurité qui en sont à la base sont mentionnés et on discute leur importance dans la réalisation d'une application uniforme ainsi que dans la conception des châteaux de transport.

Des châteaux de transport ont été construits conformément à ces règles pour certains réacteurs continentaux. La conception des châteaux de transport est soumise à des paramètres qui découlent des règlements, tels que la limitation de la température des combustibles en cas d'accident, et qui tendent à augmenter les coûts globaux de transport. En revanche des améliorations dans la conception ou dans l'utilisation des châteaux de transport, ainsi par exemple lorsqu'un même château est conçu pour desservir plusieurs réacteurs, peuvent permettre de diminuer les coûts de transport.

Les facteurs affectant le rapport poids total – charge utile du château de transport sont décrits afin d'estimer son incidence sur l'économie du transport.

Les tarifs de transport par route et par fer de combustibles irradiés sont précisés. Des pools d'assurance couvrent les risques nucléaires aux tiers, les primes sont chiffrées.

Une estimation tenant compte des coûts globaux de transport et des tarifs possibles du retraitement, pour de l'uranium naturel ou faiblement enrichi, montre qu'il serait raisonnable d'envisager une routine de transports vers des usines de traitement centrales. On montre l'influence des frais du transport sur le coût total du cycle de combustible.

A/422 Дания

Транспортировка облученного ядерного горючего в Европе

Ф. Маркус, Ж. Асие

В данном докладе описывается действующее в Европе положение о транспортирвке облу-

ченного горючего, его административные, технические и экономические аспекты и влияние на стоимость топливного цикла. Транспортировка облученного горючего, обработанного делящегося материала и, по-видимому, отходов, осуществляемая между различными странами Европы, станет в ближайшие годы обычным делом.

Начиная с 1963 года, тепловыделяющие элементы типа MTR транспортируются из континентальной части Европы для проведения переработки. Массовые перевозки облученного горючего первых энергетических реакторов еще не начаты, однако предполагается, что до 1968 года энергетические реакторы европейского континента дадут 270 *т* облученного горючего. Большая часть этого горючего имеет достаточно большую ценность, экономически оправдывающую его переработку, в связи с чем ожидается увеличение масштабов транспортировки горючего.

Приводится краткий обзор национальных законодательств, правил транспортировки и порядка получения разрешения на ее осуществление, принятых в ряде стран Европы. Рассматриваются международные правила и лежащие в их основе принципы безопасности и обсуждается важность введения единых правил транспортировки и унификации конструкции контейнеров.

Контейнеры, изготовленные в соответствии с действующими правилами, используются на ряде реакторв континента. Параметры, определяющие конструкцию контейнеров, оценивались на основе обязательных требований, таких, как ограничение температуры горючего во время аварийных условий, что привело к увеличению общих транспортных расходов. Однако в результате внесения усовершенствований в конструкцию контейнеров и в порядок их использования, например с применением контейнеров для обслуживания более чем одного реактора, можно добиться сокращения транспортных расходов. Описываются факторы, оказывающие влияние на стоимость загрузки контейнеров и учитывающие экономию, полученную при транспортировке.

Указываются тарифы на транспортировку облученного горючего по железным и грунтовым дорогам. Страховое покрытие от ядерного ущерба третьей стороне обеспечивается через страховые компании; приводятся размеры страховых премий.

Сравнение результатов расчета общих расходов на транспортировку и вероятных тарифов на переработку урана, близкого к природному, показывает, что было бы целесообразно осуществлять перевозки к удобно расположенным установкам по переработке. Определен вклад расходов по транспортировке в общую стоимость топливного цикла.

A/422 Dinamarca

Transporte de combustibles nucleares irradiados en el continente europeo

por F. Marcus y J. Asyee

Esta memoria hace una descripción general de la situación actual del transporte de combustibles irradiados en el continente europeo, de sus aspectos administrativos, técnicos y económicos, y de su influencia sobre el coste del ciclo del combustible. Dentro de unos pocos años llegará a ser una rutina el transporte, entre diferentes países, de combustibles irradiados, de material fisible tratado y posiblemente de residuos radiactivos.

Desde el año 1963 se han transportado de manera normal desde el continente, elementos combustibles de tipo MTR para su tratamiento. Los envíos importantes de los primeros reactores de potencia no han comenzado aún pero se prevé que serán descargadas 270 toneladas de combustible irradiado de los reactores de potencia del continente antes de 1968. Una parte importante de este combustible es de un valor suficientemente elevado como para justificar económicamente su tratamiento y por consiguiente se prevé un aumento de los envíos de combustible.

Se hace un breve estudio de las legislaciones nacionales y de los reglamentos de transporte así como de los procedimientos administrativos necesarios para la autorización del transporte, tal como está establecido en varios países europeos. Se mencionan las reglamentaciones internacionales y sus principios de seguridad inherentes y se discute su importancia para obtener una aplicación uniforme y unas características comunes de los recipientes blindados de transporte.

Se han fabricado recipientes blindados, de acuerdo con las reglamentaciones, para cierto número de reactores continentales. Los parámetros que determinan el proyecto del recipiente proceden de los requisitos obligatorios tales como la limitación de la temperatura del combustible en caso de accidente, lo cual tiende a aumentar los costes globales del transporte. Por otra parte, las mejoras en el proyecto o en el coeficiente de utilización del recipiente, por ejemplo, en el caso de que se haya proyectado para servir a más de un reactor, puede lograr una reducción en los costes del transporte. Se describen los factores que afectan la carga útil del recipiente con vistas a su efecto sobre la economía del transporte.

Se indican tarifas para el transporte del combustible irradiado por ferrocarril y carretera. Se dispone de seguro contra daños nucleares a tercero a través de asociaciones de compañías de seguros; se citan primas indicativas.

Una estimación de los costes globales del transporte junto con las tarifas posibles del tratamiento para uranio casi natural, indica que podrían ser razonables los envíos normales a las centrales de tratamiento. Se indica la influencia de los costes del transporte sobre el coste del ciclo total del combustible.

Containers de transport pour éléments combustibles irradiés des réacteurs de recherche BR2 et HFR

par A. de Streel,* J. Van Mulders,* T. de Schrynmakers de Dormael, G. Bonnet,*** C. Garric *** et M. Trousson *****

Le Bureau d'études nucléaires (BEN) et le Bureau d'études industrielles Fernand Courtoy (BEI) ont effectué récemment sous contrat et en collaboration avec l'EURATOM l'étude de 2 containers blindés destinés au transport des éléments de combustible irradiés des réacteurs de recherche HFR et BR2. Ces containers sont conçus pour répondre aux critères de sécurité relativement sévères imposés par les règlements USAEC 10CFR-72, *Protection Against Radiation in the Shipment of Irradiated Fuel Elements*, AIEA et RID. Le container HFR a déjà été approuvé par l'USAEC et est en instance de l'être par l'autorité compétente du Royaume-Uni; le container BR2, conçu suivant les mêmes principes, et en cours d'examen.

La firme Edlow Lead Co., chargée de la présentation des rapports à la USAEC a revu certains calculs et apporté quelques modifications aux projets, principalement dans le but de simplifier le dessin et de réduire les frais de fabrication.

DESCRIPTION DES CONTAINERS ET DE LEURS ACCESSOIRES

Les containers HFR et BR2, représentés aux fig. 1 et 2, sont destinés à transporter respectivement 16 éléments HFR du type MTR et 13 éléments BR2 formés de tubes concentriques. Ils sont conçus pour un refroidissement externe par convection naturelle dans l'air avec ailettes verticales et ont été initialement prévus pour un refroidissement interne par circulation naturelle d'eau, quoiqu'on envisage actuellement de transporter les éléments à sec. Un panier interne contenant les éléments combustibles est destiné à faciliter les opérations de manutention et de déchargement. Le poids total d'un container est d'environ 13 t. Un châssis d'arrimage et de protection contre le feu d'un poids d'environ 5 t a été étudié pour chaque container. Les caractéristiques principales de conception et de construction sont détaillées ci-après.

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Containers

Dans le but d'améliorer la résistance structurelle, la forme des containers a été déterminée de telle sorte que la majorité des accessoires et les tourillons soient reportés à la partie supérieure où se trouve une enceinte de protection. Les avantages de cette disposition sont les suivants:

a) Eviter toute partie saillante dépassant le gabarit de la surface externe en plaçant les tourillons et la plupart des accessoires à un endroit où ils ne peuvent perforer la tôle extérieure de rétention du plomb;

b) Assurer la protection du bouchon au fond d'une cavité, de façon à le soustraire aux chocs directs et à réduire les possibilités de fuite;

c) Permettre de monter la majorité des accessoires tels que vanne, manomètre, thermomètre, disque de rupture, etc., à l'intérieur de cette cavité et de les protéger ainsi contre les arrachements et détériorations possibles;

d) Permettre le guidage du bouchon lors de sa manipulation sous eau.

Afin de faciliter la décontamination, toutes les parties structurelles des containers sont réalisées en acier inoxydable. L'acier NSP3 (0,07% C—15% Cr—7,2% Ni—2,3% Mo—1,1% Al) des Aciéries d'Ugine a été retenu par suite de ses caractéristiques principales suivantes, favorables tant du point de vue résistance à la corrosion que du point de vue mécanique: haute limite élastique, haute résistance à la rupture et coefficient de dilatation élevé pour réduire les tensions dues aux dilatations différentielles entre le plomb et l'acier. Cet acier se prête également bien à la coulée de la partie supérieure.

Afin de favoriser le transfert de chaleur, on réalise une liaison entre le plomb et les parois d'acier latérales des enveloppes interne et externe. D'après les renseignements reçus de différents constructeurs, il est possible d'obtenir, sur un pourcentage important de la surface, une liaison métallurgique de caractéristiques de rupture au moins égales à celle du plomb, entre le

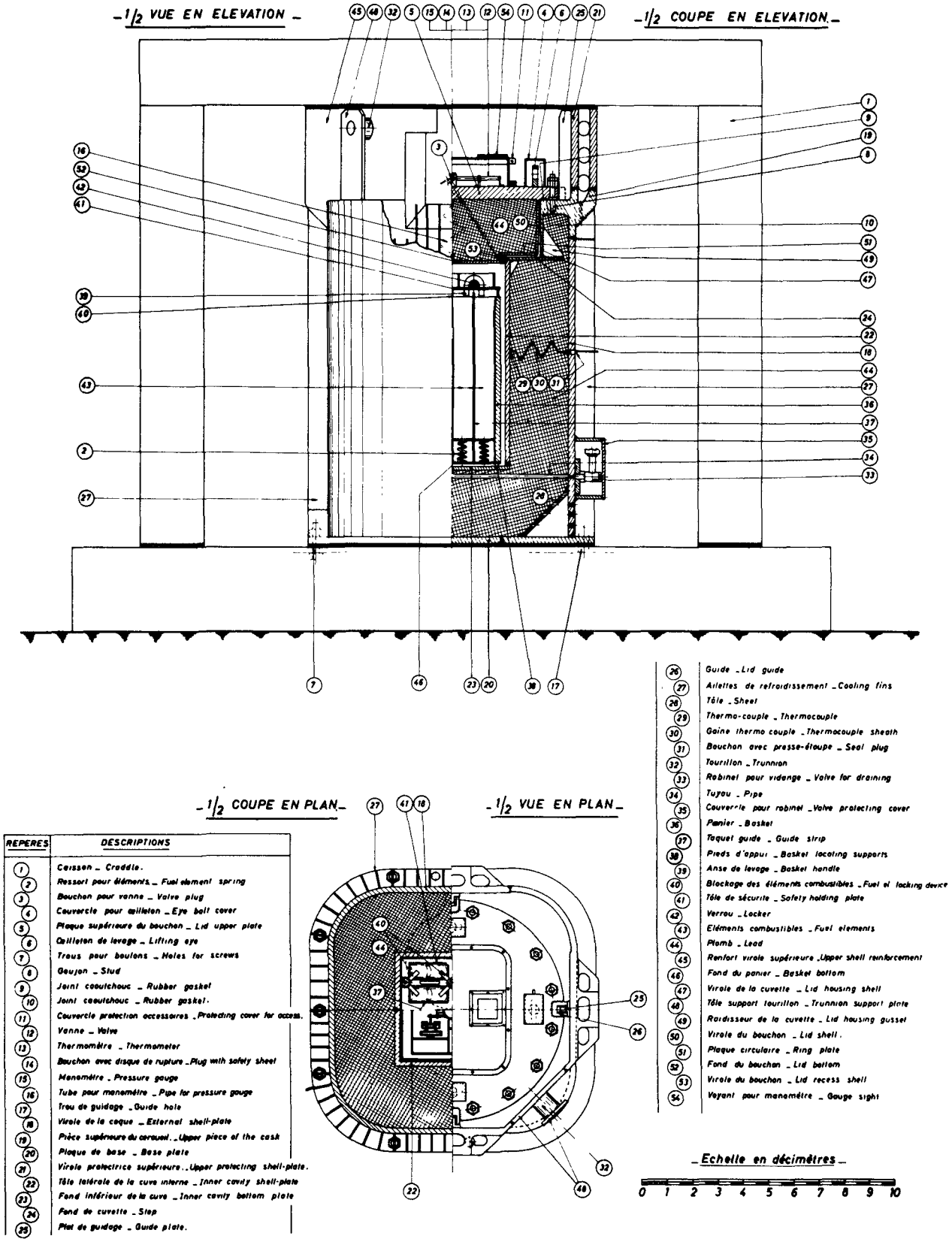
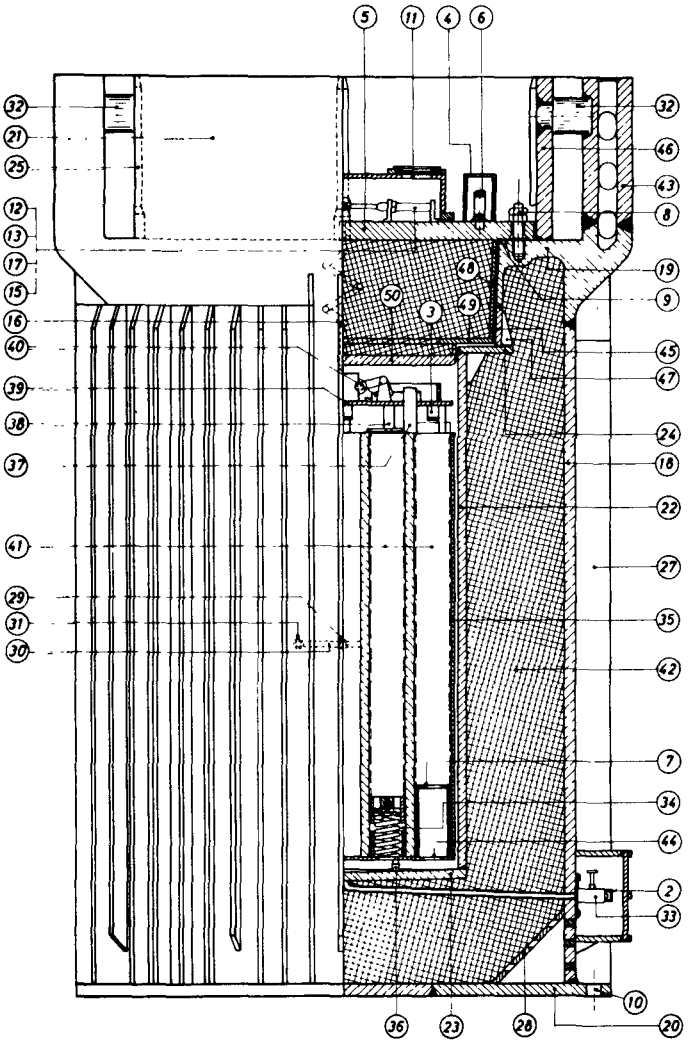
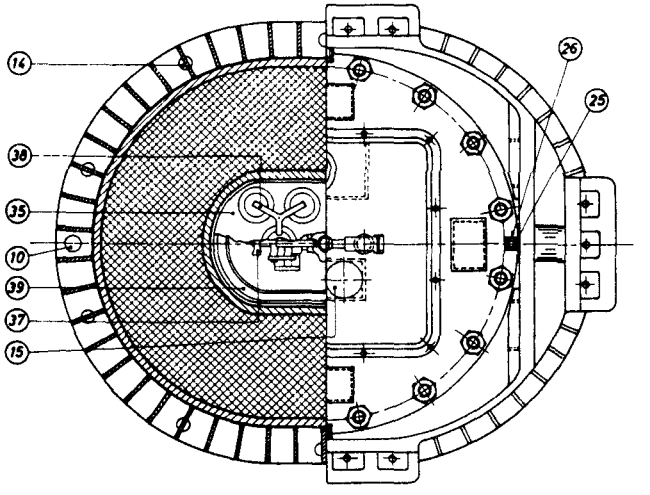


Figure 1. Vue générale du container HFR



1/2 Coupe en plan 1/2 Vue en plan



Légende

REP	DESCRIPTIONS	
2	Bouchon de vanne	Valve plug
3	Bossage tôle de sécurité	Safety holding plate boss
4	Couvercle pour oeilleton	Eye bolt cover
5	Plaque supᴱᵉ du bouchon	Lid upper plate
6	Oeilleton de levage	Lifting eye
7	Anneau d'appui	Holding ring
8	Goujon	Stud
9	Joints caoutchouc	Rubber gaskets
10	Trou pour pied de centrage	Hole for guide pin
11	Couvercle protect des access	Protecting cover for access.
12	Vanne	Valve
13	Manomètre	Pressure gauge
14	Trou pour boulon	Hole for bolt
15	Thermomètre	Thermometer
16	Tube vers manomètre	Pipe to pressure gauge
17	Disque de rupture	Rupture sheet
18	Virole de la coque	External shell plate
19	Piece supᴱᵉ du cercueil	Upper piece of the cask
20	Plaque de base	Base plate
21	Virole protectrice supérieure	Upper protecting shell plate
22	Tôle latérale de la cuve interne	Inner cavity shell plate
23	Fond infᴱ de la cuve	Inner cavity bottom plate
24	Fond de cuvette	Step
25	Plat de guidage	Guide plate
26	Guide du bouchon	Lid guide
27	Ailettes de refroidissement	Cooling fins
28	Tôle de vide de dilatation	Sheet of expansion void
29	Thermo-couple	Thermocouple
30	Gaine thermo-couple	Thermocouple sheath
31	Bouchon avec presse-étoupe	Seal plug
32	Tourillon	Trunnion
33	Robinet de vidange	Valve for draining
34	Ressort	Spring
35	Panier	Basket
36	Pied d'appui	Basket locating support
37	Anse de levage	Basket handle
38	Blocage des élem. combustibles	Fuel element locking device
39	Tôle de sécurité	Safety holding plate
40	Verrou	Locker
41	Eléments combustibles	Fuel elements
42	Plomb	Lead
43	Renfort virole supérieure	Upper shell reinforcement
44	Fond du panier	Basket bottom
45	Virole de la cuvette	Lid housing shell
46	Tôle support tourillon	Trunnion support plate
47	Raidisseur cuvette	Lid housing gusset
48	Virole du bouchon	Lid shell
49	Plaque circulaire	Ring plate
50	Fond du bouchon	Lid bottom

Echelle en décimètres

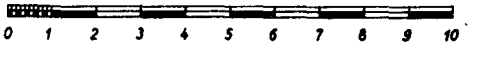


Figure 2. Vue générale du container BR2

plomb et l'acier inoxydable utilisé, et de plus la coulée du plomb peut être effectuée sans apparition de tensions appréciables lors du refroidissement jusqu'à une température de 100 °C. En-dessous de cette température, les déformations et tensions dans le plomb et l'acier, résultant des dilatations différentielles, ont été déterminées dans la gamme des températures de fonctionnement pour s'assurer qu'elles restent dans des limites admissibles.

Pour résister au feu standard, des espaces de dilatation pour le plomb, de section triangulaire avec tôles perforées, ont été prévus au bas des containers à un endroit où ils n'affectent pas l'efficacité du blindage.

La soudure sur les deux faces des ailettes de refroidissement est adoptée pour faciliter la décontamination de la surface extérieure.

Deux gaines avec thermocouples sont destinées à mesurer la température de la paroi interne. Afin d'éviter une perte de plomb en cas de feu, on réalise une double étanchéité en les soudant à l'endroit de leur pénétration des deux côtés de l'enveloppe extérieure.

Bouchons

Chaque bouchon est constitué d'un assemblage soudé en acier inoxydable NSP3 contenant un blindage de plomb. L'espace réservé à la dilatation du plomb est obtenu en laissant la solidification s'effectuer après coulée sans ajout ultérieur. Le bouchon est de forme circulaire pour faciliter l'usinage et conique pour faciliter la mise en place. Il contient une gaine inclinée contenant un thermomètre indicateur pour mesurer la température du haut de la cavité interne.

Le guidage du bouchon lors de sa mise en place sous eau s'effectue au moyen de 4 guides plats de forme appropriée fixés à la virole de protection supérieure et s'engageant dans 4 rainures à 90 ° de la plaque supérieure du bouchon.

L'étanchéité est assurée par 2 joints toriques en caoutchouc maintenus en place dans des gorges usinées dans la plaque supérieure à l'emplacement où le niveau de rayonnement gamma et la température sont minimaux. Une prise de pression obturable entre les 2 joints permet de tester individuellement chacun d'eux.

Paniers

Chaque container est muni d'un panier comportant des alvéoles destinés à contenir les éléments combustibles dont les extrémités d'aluminium ont été préalablement sectionnées pour réduire les frais de transport et de retraitement. Chaque panier est réalisé par coulée en acier inoxydable type ASTM 304 avec parois massives de l'ordre de 2 cm d'épaisseur. Le panier remplit les fonctions suivantes:

a) Servir de poison neutrophage pour réduire k_{eff} en dessous de 0,9;

b) Assurer le transfert de chaleur en l'absence d'eau de refroidissement, par conduction au sein des parois et par rayonnement des surfaces;

c) Faciliter les opérations de manutention et permettre des modifications éventuelles ultérieures;

d) Faciliter les opérations éventuelles de décontamination.

Chaque panier est muni à sa partie supérieure d'anses de manutention et de dispositifs de blocage des éléments combustibles. Les dispositifs de blocage, constitués de pales rotatives et verrouillables, ont pour but de maintenir la majorité des éléments et d'empêcher la formation d'une masse critique en cas de renversement accidentel d'un panier. Des dispositifs à ressorts sont prévus à la partie inférieure des alvéoles pour maintenir les éléments en place et admettre différentes longueurs de coupure. Une plaque de sécurité additionnelle, verrouillée au sommet du panier, a pour but de maintenir la partie active des éléments au droit des cloisons neutrophages, lors des sollicitations accidentelles de chute auxquelles le container pourrait être soumis.

Châssis d'arrimage et de protection

Les containers tels que décrits répondent aux critères de sécurité imposés par le règlement AEC 10CFR-72. Cependant, pour répondre aux recommandations de l'UKAEA, un châssis d'arrimage et de protection contre l'incendie a été mis au point. Ce châssis permet de porter de 15 à 30 pieds la hauteur de chute et maintient la température du plomb en dessous de sa température de fusion à la fin du feu standard défini par l'UKAEA (température constante de 800° C $\pm$ 20 °C pendant ½ heure). Le châssis est constitué de poutrelles croisées pour assurer la résistance mécanique et de parois en tôle revêtues d'isolant thermique contre l'incendie. Il comprend 2 parties: une plate-forme sur laquelle est fixé le container et qui assure l'arrimage au véhicule de transport, et une cloche qui porte les orifices d'entrée et de sortie de l'air de refroidissement et leur système de fermeture automatique en cas d'incendie. Le système de fermeture est constitué d'une plaque coulissante tirée par un ressort et maintenue en position dans les conditions normales de transport par une soudure à l'autre extrémité. La soudure est protégée contre les chocs par un bouclier. En cas d'incendie, la soudure fond entre 150 et 200 °C et libère la plaque qui, sous l'action du ressort, vient obturer les orifices d'entrée ou de sortie d'air. La réouverture des orifices d'air après l'incendie s'effectue en replaçant manuellement les plaques d'obturation dans leur position primitive ou, en cas de difficulté, en défonçant des parties minces avec une barre de fer.

INTÉGRITÉ STRUCTURELLE

Le calcul de la résistance de containers se heurte à des difficultés considérables qui ne pourront être élucidées que lorsque les résultats d'un grand nombre d'expériences seront disponibles.

Sollicitations normales

La complexité de la structure amène à assimiler pour chacun des calculs, partie ou totalité du container à une configuration simple en y appliquant des hypothèses raisonnablement pessimistes.

C'est ainsi que pour répondre au § 72.32a du règlement USAEC 10CFR-72, la coque extérieure a été calculée pour résister à elle seule à 10 fois le poids total du container, plomb compris. Par analogie, on considère le container comme une poutre chargée uniformément pour le calcul des tensions de flexion et comme une enveloppe mince pour le calcul des efforts tranchants au droit des appuis. A titre de vérification, négligeant l'adhérence du plomb et de l'acier, on vérifie que le flambage des parois latérales et de la plaque de fond ne peut pas se produire.

De plus, les tensions engendrées dans les boulons du bouchon sont évaluées en résolvant le système des équations de déformation de la structure supérieure du container et du bouchon. A titre de vérification supplémentaire, il a été démontré que le flambage de la bride du bouchon ne se produit pas.

Afin d'éliminer les difficultés et les incertitudes particulières aux calculs dans le domaine élasto-plastique, les épaisseurs de métal ont partout été choisies afin de ne pas dépasser la limite élastique de l'acier spécial NSP3 utilisé.

Sollicitations accidentelles

Les sollicitations qui servent de base au calcul de l'intégrité structurelle du container en cas d'accident sont décrites au § 72.32b en ce qui concerne la chute de 15 pieds, au § 72.34a en ce qui concerne la perforation et au § 73.34b en ce qui concerne l'exposition au feu standard de 1 h.

La résistance du container à une chute de 15 pieds sur l'une quelconque de ses faces a été étudiée par différentes méthodes.

En ce qui concerne l'impact sur une face latérale du container, on constate que, si on néglige l'influence du plomb, le container n'a qu'une faible résistance transversale. Au moment de l'impact, la partie centrale située entre le fond et la structure supérieure a tendance à s'aplatir tandis que les extrémités se déforment beaucoup moins. Ainsi l'énergie d'impact est en grande partie dissipée par le travail de déformation d'une poutre équivalente dont les points d'appuis sont simulés par le fond et la structure supérieure.

Une hypothèse pessimiste consiste à supposer que toute l'énergie est absorbée par le travail de déformation de la coque extérieure assimilée à une poutre sur deux appuis et à négliger la contribution du plomb à la résistance sous impact du container.

En principe, la force au moment de l'impact peut être assimilée, conformément au § 72.32b, à 60 fois le poids du container. Une deuxième hypothèse pessimiste consiste à localiser cette force au centre de la poutre. Cette force est appliquée pendant une durée non inférieure à 16 ms. Les déformations sont calculées en tenant compte de cette durée d'impact.

Une troisième hypothèse pessimiste consiste à calculer les tensions à partir de ces déformations en utilisant les équations applicables à une charge statique, c'est-à-dire, de durée d'application illimitée.

Les différentes méthodes utilisées sont extraites des publications de Hohenemser et Prager [1], Lee et Symonds [2], Symonds [3], Melvin, Greaves et Davis [4], et enfin de Timoshenko [5]. Toutes conduisent à la conclusion que les tensions de déformation engendrées ne dépassent pas la limite élastique de l'acier inoxydable NSP3 utilisé.

En ce qui concerne l'impact sur le fond ou sur le bouchon, on utilise la formule de Saint-Venant [6] pour déterminer les tensions et on vérifie en outre que les parois latérales ne flambent pas.

En ce qui concerne la perforation, l'épaisseur de l'enveloppe extérieure est supérieure à la valeur minimale requise pour satisfaire le § 72.34a.

La résistance à un incendie a été étudiée en supposant le container soumis dans un four à un feu standard de 1 h, dont l'évolution de la température est décrite à l'annexe A du règlement 10CFR-72.

Dans ces conditions, la quasi-totalité du plomb est fondue et, à cause de la valeur finie de son coefficient de diffusion thermique, les couches périphériques du blindage de plomb sont portées à des températures nettement supérieures au point de fusion. Il est donc possible, dans les limites de la précision du calcul de la distribution des températures dans le plomb, de déterminer le volume d'expansion nécessaire.

On trouve le volume disponible correspondant en considérant:

a) Les vides résultant de la différence entre la densité réelle du plomb après solidification et sa densité théorique;

b) La dilatation thermique de l'enveloppe extérieure en acier;

c) La déformation élastique de cette enveloppe sous les contraintes dues à l'expansion du plomb;

d) Les espaces d'expansion situés à la partie inférieure du container.

L'enveloppe extérieure atteint après 1 h d'exposition au feu une température voisine de 1 000° F (538° C).

Cette enveloppe est d'autre part sollicitée par la pression hydrostatique du plomb fondu et par les contraintes dues à l'expansion de celui-ci.

A la température atteinte, l'acier de l'enveloppe conserve ses propriétés et permet de résister à ces sollicitations. Le plomb ne s'échappera donc pas de l'enveloppe.

En conclusion des études d'intégrité structurelle, on constate que si des déformations permanentes se produisent, celles-ci restent toujours suffisamment faibles pour ne pas compromettre l'intégrité du blindage de plomb.

BLINDAGE

Dans le but de faciliter la manutention et d'abaisser les frais de transport, il est utile d'effectuer des calculs de blindage détaillés et précis, afin de réduire dans toute la mesure du possible les épaisseurs des parois et donc le poids des containers.

La source de rayonnement gamma peut être déterminée, soit suivant Blomeke et Todd [7] qui considère 4 groupes d'énergie, soit suivant Perkins et King [8] qui sépare l'énergie gamma en 7 groupes. Cette dernière référence permet des calculs plus précis par suite de la décomposition plus fine du spectre d'énergie gamma. Les coefficients d'atténuation et de *build-up* sont bien connus et ne diffèrent que très peu suivant la référence utilisée dans la littérature. De façon générale, pour les containers destinés au transport d'éléments combustibles irradiés, la condition limitant l'intensité de dose à 10 mR/h à 1 m de la surface accessible extérieure détermine l'épaisseur du blindage; pour cette épaisseur, on constate que l'intensité de dose à la surface est inférieure à 200 mR/h. La limite de 200 mR/h à la surface permet donc d'inclure la contribution de rayonnements localisés dus aux hétérogénéités du blindage, fentes et pénétrations de tuyauteries. En tenant compte de l'atténuation des parois d'acier, l'épaisseur du blindage de plomb des containers varie entre 22 et 25 cm suivant la paroi considérée. En cas d'accident, l'intensité de dose admissible à 1 m de la surface extérieure est de 1 R/h. A cette dose, 100 fois plus forte que dans les conditions normales de transport, correspond généralement une réduction de l'épaisseur de plomb d'environ 9 cm, le facteur d'atténuation pour une différence de 1 cm de plomb étant d'environ 1,7. On constate donc que l'intensité de dose résultant d'un enfoncement maximal de 3 pouces (7,62 cm), correspondant aux conditions accidentelles de chute considérées dans le règlement 10CFR-72, ne dépassera pas la limite admissible, même si cet enfoncement se produit entièrement dans le blindage de plomb. Ce n'est que pour le cas du feu standard, en cas de fusion du plomb, remplissage des espaces de dilatation et resolidification ultérieure, qu'il y a lieu de vérifier, pour toutes les positions possibles du container, que l'épaisseur de blindage de l'ensemble d'une paroi n'est pas réduite de plus de 9 cm environ.

Le rayonnement de fuite à travers les fentes du bouchon et les tuyauteries de pénétration a été étudié en détail. Comme ces fentes et ces tyuauteries présentent des coudes, on a d'une part une composante due aux réflexions et d'autre part une composante directe due à la réduction d'épaisseur de blindage à l'endroit de ces pénétrations. Dans chaque cas, un facteur géométrique d'atténuation est applicable par suite des dimensions relativement réduites des pénétrations. La composante réfléchie se calcule par la section différentielle de Klein-Nishina en fonction de l'énergie de rayonnement, de l'angle de réflexion et de la quantité de matière réfléchissante. Contrairement à l'atténuation du blindage qui s'accroît lorsque l'énergie gamma décroît, la dose réfléchie augmente lorsque l'énergie gamma diminue. Cependant, en tenant compte des facteurs géométriques d'atténuation, la composante réfléchie est généralement peu importante, surtout pour des réflexions successives. Par contre la composante directe, atténuée par une épaisseur de blindage moindre et par la forme géométrique de la pénétration, est généralement plus importante et intervient dans le dessin des pénétrations de la manière suivante: i) pour les tuyauteries, il y a lieu de limiter la longueur des tronçons dont l'axe se confond avec un rayon direct perpendiculaire au blindage entre la source et l'observateur; ii) la fente du bouchon dirigée vers un observateur extérieur doit se trouver face à un point du blindage où le flux gamma est déjà suffisamment atténué. Pour cette raison, un ressaut dans le bouchon a été prévu à une certaine distance de la cavité interne.

La présence des tronçons de tuyauterie situés dans le blindage parallèlement aux faces du container revient à diminuer localement l'épaisseur du blindage de plomb. Comme ces tronçons ne recouvrent qu'une très faible fraction de la surface totale de la source vue par l'observateur, l'accroissement de dose qui y correspond est relativement faible.

CRITICITÉ

Chaque container est destiné à transporter un nombre d'éléments qui dépasse le nombre critique. Il y a donc lieu de prévoir des parois neutrophages pour réduire le k_{eff} de la charge à moins de 0,9. Pour faciliter la décontamination, des parois d'acier inoxydable sont favorables. Comme, pour des raisons de transfert de chaleur, le panier doit être massif avec des parois d'une certaine épaisseur, cet acier peut également servir de poison neutrophage avec l'avantage qu'il peut être considéré comme indestructible dans les conditions accidentelles envisagées. Le cofficient de multiplication effectif calculé vaut respectivement 0,69 pour le container HFR et 0,74 pour le container BR2.

Les calculs de criticité ont été effectués de façon normale en considérant différentes régions homogénéisées et en utilisant 4 groupes d'énergie dont 3 groupes rapides et 1 groupe thermique. Le panier HFR a été étudié en géométrie rectangulaire à 2 dimensions, tandis que la géométrie cylindrique a été retenue pour l'étude BR2 par suite de la forme des éléments et de leur panier. Il a également été vérifié que le nombre de cellules unitaires du panier était de loin inférieur à 0,75 fois le nombre critique de ces cellules. Les codes et la calculatrice numérique Ferranti Mercury du Centre d'étude de l'énergie nucléaire à Mol ont été utilisés pour effectuer les calculs.

Les méthodes de calcul ont été vérifiées au moyen de résultats d'expériences critiques effectuées respectivement – pour les éléments HFR, sur des éléments quasi identiques du type MTR [9]; pour les éléments BR2, sur des éléments semblables du type PTR [10].

Les vérifications ont montré que les méthodes utilisées permettent de calculer k_{eff} avec une précision meilleure que 1 % dans le sens de la sécurité (surestimation de k_{eff}).

L'interaction des paniers manipulés sous eau dans les piscines peut être calculée de la même façon que la sous-criticité d'un panier pour déterminer le nombre de paniers que l'on peut placer côte à côte sans que k_{eff} dépasse 9,9, Cependant, ces calculs ne sont pas indispensables si on prend la précaution de laisser une distance d'eau de 30 cm (1 ft) entre paniers pour supprimer toute interaction entre ceux-ci.

L'interaction entre plusieurs containers placés côte à côte a été analysée conformément aux critères de sécurité relatifs aux angles d'interaction donnés à la référence [11]. L'étude a montré qu'il n'y avait pas de problème d'interaction, quel que soit le nombre de containers chargés placés côte à côte.

La quantité totale d'uranium 235 transportée (2,880 kg pour le container HFR et 3,172 kg pour le container BR2) est de loin inférieure à la masse critique minimale d'uranium 235 métallique pur ou dilué dans un métal quel que soit le réflecteur [12]. La fusion totale du combustible, qui n'est cependant pas à craindre dans les conditions accidentelles envisagées par les règlements, ne peut donc créer de masse critique.

ÉVACUATION DE LA CHALEUR

La chaleur résiduelle des éléments combustibles est produite par absorption des rayonnements bêta et gamma des produits de fission dans les matériaux de structure du container.

Cette chaleur dépend des conditions d'irradiation des éléments combustibles, des facteurs de canal chaud liés à la position des éléments dans le réacteur et enfin de la période de refroidissement précédant le chargement du container.

L'évaluation de la chaleur résiduelle a été effectuée en utilisant les données de Blomeke et Todd [7] et de Perkins et King [8].

La presque totalité du rayonnement gamma est absorbée à l'intérieur du panier en acier inoxydable, alors que la totalité du rayonnement bêta est absorbée dans les plaques des éléments combustibles.

Pour calculer les températures atteintes dans le container, deux cas ont été envisagés:

a) Le réfrigérant interne est de l'eau (transport humide);

b) En l'absence d'eau, le réfrigérant interne est supposé être de l'air (transport à sec).

Tous les calculs ont été menés dans le but de répondre aux exigences du § 72.38 du règlement 10CFR-72.

Transfert de la chaleur au milieu ambiant

Les conditions d'ambiance sont définies par le § 72.38c: air calme à 100 °F (37,8 °C) et insolation maximale compatible avec le trajet suivi par le container. D'autre part, d'après le § 72.38a (1), la température de toute partie facilement accessible de la surface extérieure du container ne peut dépasser 180 ° F (82,2°C).

Si on désire obtenir le refroidissement du container par des moyens naturels (absence de dispositif mécanique de refroidissement), on est amené, pour assurer le transfert de la chaleur vers le milieu ambiant, à prévoir des ailettes soudées à l'enveloppe extérieure du container. Le nombre et les dimensions de ces ailettes sont calculés en tenant compte des deux modes d'échanges calorifiques suivants:

a) *Convection naturelle de l'air.* La capacité de transfert des ailettes dépend de leur rendement. Celui-ci est égal au produit du rendement d'une ailette supposée isolée, fonction de la conductivité du matériau constitutif de l'ailette, d'un facteur fonction de l'interaction due à la proximité de deux ailettes adjacentes, et d'un facteur correctif expérimental [13].

b) *Rayonnement vers le milieu ambiant.* L'émissivité de la paroi extérieure, constituée par de l'acier inoxydable soumis à de fréquents lavages et brossages pour décontamination, a une valeur relativement basse à la température ambiante prescrite par le règlement. Cependant l'émissivité apparente de la surface extérieure du container est plus élevée grâce au rayonnement accru des cavités formées par 2 ailettes adjacentes [14].

Chute de température à travers les parois

Les parois du container se composent de deux enveloppes externe et interne en acier entre lesquelles se trouve le plomb formant blindage. Le calcul du transfert de la chaleur à travers ces parois, basé sur la

conductibilité de chacune des couches traversées, est classique à cela près qu'il n'est pas possible de garantir une liaison parfaite entre le plomb et l'acier. On suppose, dans les calculs thermiques des containers étudiés, que cette liaison est parfaite pour une moitié de la surface de contact et que, au moment de la coulée, le retrait du plomb provoque pour l'autre moitié de cette surface un vide rempli d'air dont l'épaisseur est évaluée à environ 1 mm.

On constate avec une bonne approximation que, dans ces conditions, la chute de température dans les parois est environ égale au double de la chute de température dans le seul blindage de plomb.

Chutes de température dans la cavité interne du container

Refroidissement par eau

Les éléments combustibles, ainsi qu'il a été décrit plus haut, sont contenus dans les alvéoles d'un panier en acier inoxydable qui lui-même est logé à l'intérieur du container. L'eau en s'échauffant monte le long des éléments combustibles et se refroidit en descendant dans l'interstice existant entre le panier et l'enveloppe interne. La circulation naturelle qui en résulte a été calculée théoriquement. Environ 90% de la chaleur résiduelle des éléments combustibles est transférée par la circulation naturelle, tandis que les 10% restants sont extraits par conductivité.

Refroidissement par air

Dans le cas d'absence d'eau, la contribution de la circulation naturelle de l'air à l'intérieur du container est négligeable. La chaleur résiduelle est transférée à travers les parois métalliques des éléments combustibles et du panier, et les espaces d'air séparant les éléments du panier et le panier de la paroi interne du container. La chute de température la plus importante, environ 350 °F (195 °C), se produit dans ce dernier espace d'air de 5 mm d'épaisseur. Approximativement une moitié de la chaleur y est transférée par conduction et l'autre moitié par rayonnement.

Températures maximales atteintes

Pour les containers BR2 et HFR exposés à de l'air ambiant à 100 °F (37,8 °C) et à l'insolation maximale, la température des éléments combustibles atteint les valeurs suivantes:

Réfrigérant	BR2	HFR
Eau	215 °F (102 °C)	223 °F (106 °C)
Air	672 °F (356 °C)	797 °F (425 °C)

Dans le cas de l'eau, il en résulte une légère autopressurisation des containers qui sont conçus pour la pression maximale admise par le règlement 10CFR-72.

Dans le cas de l'air et pour une température d'avarie égale à la tempréature de fusion de l'aluminium (1 150 °F, 621 °C), les températures atteintes respectent la marge de 300 °F (167 °C) en-dessous de la température d'avarie.

On peut donc conclure que le transport à sec envisagé actuellement pour les deux containers HFR et BR2 reste conforme au règlement 10CFR-72.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/448 Belgium

Shipping casks for the BR2 and HFR research reactors

By A. de Streel *et al.*

The transport of irradiated fuel elements creates many problems. The technique involved is of recent origin, and the calculation, construction and testing methods and procedures have not all been borne out by experience yet, but are still capable of improvement. Moreover, the design of the casks must comply with the relatively severe criteria prescribed in the official regulations, some of which are still being drafted, or are subject to amendment. The main purpose of these regulations is to protect the public from radiation hazards.

A study of casks for the shipment of spent fuel elements from the BR2 and HFR research reactors, recently carried out by the Bureau d'études nucléaires (BEN) and the Bureau d'études industrielles Fernand Courtoy (BEI) under contract, and in co-operation with EURATOM, is described. The HFR cask has already received USAEC approval and is on the point of being accepted by the United Kingdom Authority; the BR2 cask, designed on the same principles, is at present under examination.

The paper summarises the solutions chosen for the various theoretical and technological problems in order to comply, under normal and emergency conditions of transport, with the safety requirements laid down in the new regulations in force, mainly the AEC 10 CFR Part 72 and the AIEA regulations.

A description with drawings is given; and the technological fabrication problems, the design features and layout chosen to guarantee the strength of the casks, and their handling, anchorage, structural soundness and shielding are reviewed.

Each cask contains a quantity of fuel elements which exceeds the critical number. Solid, stainless steel baskets, which act as a neutron poison, are therefore used to fulfil the sub-criticality requirements. They allow easier decontamination, may be considered as indestructible in the accident conditions envisaged, and act as an internal heat-transfer medium.

The problem of residual-heat removal, to maintain, for safety reasons, an adequate margin below the fuel-element-failure temperature, is investigated theoretically. The internal primary coolant can be either water or air, and fins are provided for external cooling.

The resistance of the cask to fire is investigated from the following points of view: critical temperatures, mechanical strength and shielding efficiency. A fire-protection frame with insulating walls has been devised. This frame, which surrounds the cask, is also useful for shock-absorbing and anchoring purposes.

A/448 Бельгия

Транспортные контейнеры для облученных тепловыделяющих элементов исследовательских реакторов BR-2 и HFR

А. де Штрель *et al.*

Транспортировка облученных тепловыделяющих элементов связана с многими проблемами, возникающими в силу следующих причин. Прежде всего речь идет о новой технике, методы расчета, конструирования и испытаний которой еще не апробированы практикой и могут быть усовершенствованы. Кроме того, необходимо соблюдать довольно жесткие, предписанные официальными постановлениями нормы, часть из которых еще разрабатывается или исправляется. Эти нормы преследуют прежде всего задачу обеспечения безопасности населения.

В докладе приводятся результаты изучения транспортных контейнеров для облученных тепловыдляющих элементов исследовательских реакторов BR-2 и HFR, проведенного недавно сотрудниками Отдела ядерных исследований и Отдела промышленных исследований в соответствии с соглашением и в сотрудничестве с Евратомом. Контейнер для твэлов реактора HFR уже одобрен КАЭ США и в настоящее время находится на рассмотрении английского компетентного органа; контейнер для твэлов реактора BR-2, представляющий собой прототип контейнера для твэлов реактора HFR, в настоящее время находится в стадии изучения.

В докладе вкратце излагаются решения, принятые по ряду теоретических и технологических проблем применительно к обычным и непредвиденным условиям перевозок и в духе соблюдения норм безопасности, предписанных новыми постановлениями, вошедшими недавно в силу, в частности постановлениями AEC 10 CFR, часть 72, и AIEA.

Приводится описание контейнеров, излагаются технологические проблемы, связанные с их изготовлением, а также меры, обеспечивающие механическую прочность, крепление, целостность конструкции, защиту контейнеров и их ремонт.

Для того чтобы обеспечить подкритичность контейнеров, каждый из которых вмещает груз, превышающий критическое число тепловыделяющих элементов, в качестве поглотителя нейтронов применяют массивные сетки из нержавеющей стали. Они легко поддаются дезактивации и не подвергаются разрушению в аварийных условиях. Кроме того, они способствуют внутреннему теплообмену.

Проблема отвода остаточного тепла для создания постоянного резерва безопасности при аварийной температуре рассматривалась чисто теоретически. Внутренним охладителем может быть либо вода, либо воздух, а наружное охлаждение может быть обеспечено за счет ребристой поверхности контейнеров.

Последствия имитированного загорания, вызванного обычным огнем, определены как с точки зрения критических температур, так и с точки зрения механической прочности и эффективности защиты. Рассмотрена также возможность применения защитного каркаса с изолирующими перегородками как средства защиты от огня. Кроме того, такой каркас может служить для усиления механической прочности и надежности крепления контейнера.

A/448 Bélgica

Recipientes de transporte para los elementos combustibles irradiados en los reactores experimentales BR2 y HFR.

por A. de Streel *et al.*

El transporte de los elementos combustibles irradiados plantea muchos problemas por las siguientes razones: por un lado, se trata de una técnica reciente cuyos métodos de cálculo, de construcción y de ensayo no están todos debidamente sancionados por la experiencia y pueden aún ser mejorados; por otra parte, es preciso respetar los criterios relativamente severos impuestos por las reglamentaciones oficiales, algunas de las cuales se hallan todavía en curso de elaboración o sujetas a modificaciones. Estos criterios persiguen como fin fundamental garantizar la seguridad de las poblaciones.

Se describe el estudio de los recipientes de transporte para elementos combustibles irradiados en los reactores experimentales BR2 y HFR, realizado recientemente por BEN y BEI bajo contrato y con la colaboración de EURATOM. El recipiente HFR ya ha sido aprobado por la USAEC y está a punto de serlo por la autoridad competente del Reino Unido; el recipiente BR2, concebido con arreglo a los mismos principios, se halla sometido a examen.

La memoria resume las soluciones aportadas a los muchos problemas de índole teórica y tecnológica que han sido resueltos, para ajustarse, en condiciones normales y accidentales de transporte, a los criterios de seguridad impuestos por los nuevos reglamentos vigentes, sobre todo en lo que respecta a las reglamentaciones AEC-10-CFR, parte 72, y OIEA.

Se da una descripción que incluye planos, además de una visión de conjunto de los problemas tecnológicos relacionados con la fabricación y de los dispositivos adoptados para asegurar la resistencia mecánica, el manejo, la estiba, la integridad estructural y el blindaje.

Para garantizar la subcriticidad de los recipientes, cada uno de los cuales contiene una carga que excede del número crítico de elementos, se utilizan cestas de acero inoxidable macizo como venenos neutrófagos, las cuales son fácilmente descontaminables, pueden ser consideradas como indestructibles dentro de las condiciones accidentales previstas y, además, favorecen la transmisión del calor interno.

Desde un punto de vista teórico se examina el problema de la eliminación del calor residual de los elementos combustibles con objeto de mantener siempre un margen de seguridad suficiente con respecto a su temperatura de ruptura. El refrigerante interno puede estar constituido por agua o aire y la refrigeración externa está garantizada mediante aletas.

Se evalúan los efectos producidos en un incendio simulado por un fuego patrón, tanto desde el punto de vista de las temperaturas críticas como en lo que se refiere a la resistencia mecánica y a la eficacia del blindaje. Se ha estudiado un bastidor de protección contra el fuego, de paredes aislantes, que sirve también para mejorar la resistencia mecánica y para asegurar las amarras.

Record of Session 3.8

Chemical processing, plant safety; fuel transport

Chairman: H. N. Sethna (India)

Paper P/508 (presented by J. Tadmor)

DISCUSSION

J. A. McBRIDE (United States of America): Has application of the principles described in the paper resulted in modifications with respect to the siting, process systems, or components of any proposed installations?

J. TADMOR (Israel): The guide was prepared for the hazards evaluation study of the hot laboratory to be built at the Israeli Nuclear Centre of Nahal Soreq. Safety people realize that the constructors and operators of nuclear plants do not willingly carry out hazards evaluation work, and we thought this guide would facilitate such a study.

A. K. GANGULY (India): I should like to comment that hazards evaluations of plants under design or in operation often tend to emphasize safety in normal operation and in regard to the maximum credible accident. In between these two conditions lies the entire range of credible accidents, which do happen on occasion. The cumulative effect of such accidents over the operational lifetime of a plant should be checked from the point of view of acceptability of the plant in a given environment. Containment requirements may need to be quite exacting to ensure safety in such accidents.

J. TADMOR (Israel): I completely agree with Dr. Ganguly that in view of the higher probability of relatively small accidents, they should be analysed and evaluated as carefully as the maximum credible accident. This is in fact emphasized in the latter part of our paper. However, if safeguards are provided to cope with the whole range of accidents, for a given plant, and if the evaluation of containment, radiation dose and environmental contamination shows that a plant is acceptable on a certain site from the MCA standpoint, I think it would also be acceptable from the point of view of relatively smaller and even more frequent accidents.

Paper P/278 (presented by J. A. McBride)

DISCUSSION

A. K. GANGULY (India): Are any figures available to indicate the relative frequency of radiation injuries and injuries due to other (conventional) causes in the operation of such a plant?

J. A. McBRIDE (United States of America): I have no specific figures at hand, but radiation accidents (i.e. exposure above administratively established limits) are more infrequent than ordinary industrial accidents requiring first-aid treatment or more.

B. G. DUBOVSKY (USSR): What is the method of controlling ^{235}U concentration in processing enriched uranium?

J. A. McBRIDE (United States of America): Administratively, concentration in batch processes may be controlled by controlling the quantity of fuel charged to the dissolver, and by routine analyses of all uranium streams in critically safe vessels before transfer to unsafe systems. Dissolvers are inspected for undissolved heels before a fuel recharge. The concentration of TBP in the extraction is carefully monitored, and critical streams have flow alarms to signal an interruption of flow, which might otherwise lead to an unsafe increase in uranium concentration. Neutron monitors may also be used to warn that uranium concentrations are increasing towards the critical point. In this connection I would refer you to Table 2 of the paper.

B. G. DUBOVSKY (USSR): Do you measure the degree of subcriticality of the chemical apparatus in processing fissionable material?

J. A. McBRIDE (United States of America): For the most part, the neutron monitors in use detect neutron flux by measuring the intensity of the (α, n) reaction; they thus indicate plutonium concentration in the streams being monitored. In a few instances, neutron multiplication is used, and in these cases, the instruments will indicate the degree of subcriticality of the system.

Paper P/280 (presented by M. Levenson)

DISCUSSION

K. D. B. JOHNSON (United Kingdom): If you had to start a similar plant design again, would you make any changes on the basis of your experience?

M. LEVENSON (United States of America): If we were to recommence our design, we would make the

air cell round, like the argon cell: the increased visibility and the opportunity to operate material-handling equipment, such as cranes and manipulators, on endless tracks is a great advantage from the point of view of avoiding shut-downs due to equipment failure.

K. D. B. JOHNSON (United Kingdom): If the facility had to be designed for 10 or 100 times the quantity of fuel, would your design philosophy be much different?

M. LEVENSON (United States of America): It is difficult to separate the changes we would make into those relating to capacity and those relating to the transition from a research facility to a production facility. The capacity of the present processing equipment could of course be expanded by operating with three shifts seven days a week instead of the present single shift, working five days a week. From the safety standpoint, I know of nothing we would change: the amount of fission products and enriched material being handled is already so great that an increase in capacity would introduce no new problems.

B. G. DUBOVSKY (USSR): What sensors are used to register neutrons in strong λ fields with intensities about 10^6 r/h, and how reliable is their performance?

M. LEVENSON (United States of America): Neutrons are not measured in the Fuel Cycle Facility of EBR-II, and we have no data on their performance in such a field.

Paper P/84 (presented by D. Breton)

DISCUSSION *

J. SCHMETS (Belgium): The papers we have just heard on critical studies show excellent agreement between calculation and experiment. Can one rely to the same extent on critical mass calculations for fissionable materials in the solid state, such as powders?

D. BRETON (France): We have no practical experience on the critical masses of solid materials and powders. With powders the problem of density has to be solved.

B. G. DUBOVSKY (USSR): It would be interesting to compare calculations by the "equivalent dimensions" method ** with the results of your experiments on the interaction of subcritical reactors.

D. BRETON (France): We have not seen a description of the method to which you refer, but as soon as we are familiar with it we shall be glad to compare results.

* See also discussion on paper P/381.

** Dubovsky, B. G., *et al.*, At. Energ. (USSR), *16* (1), 16 (1964).

Paper P/381 (presented by B. G. Dubovsky)

DISCUSSION

M. LEVENSON (United States of America): I have a question to Mr. Dubovsky on paper P/381 and to Mr. Breton on paper P/84. Do you think we are as yet able to depend on the results of calculation for determination of design parameters, e.g. for new processing installations, or do you believe it is necessary to mock up any proposed new type of installation for nuclear test prior to construction?

B. G. DUBOVSKY (USSR): In the case of vessels of regular geometry (spheres, cylinders, etc.), the size of the permissible charge can be calculated with sufficient accuracy (about 10%), and it is enough to conduct spot checks for different uranium or plutonium concentrations.

In the case of irregular geometries (cylinders with conical or hemispherical base etc.), more caution should be exercised in relying on calculations: control experiments should be conducted for each vessel geometry, and the results should be compared with buckling calculations. In this connection, I would refer you to an article by V. K. Bogatyrev. ***

D. BRETON (France): The question is rather intricate. In the CEA programme we have conducted experiments on elements of simple geometry to check the validity of methods and constants. We reserved the Monte Carlo type calculations for studies relating to elements of complicated geometry. In view of the speed of computers we think this method can often be used, with a consequent reduction in the number of experiments that have to be carried out.

Paper P/422 (presented by F. R. Marcus)

DISCUSSION

E. IANSITI (Italy): Your paper states that the Italian shipment from Ispra to Casaccia did not meet the approved standards. This is not quite exact. The shipment was not only in accordance with the draft United States regulation in 10.CFR Part 72 but was so conceived that it could withstand the 30 ft drop followed by fire. Although designed two years ago, the container conformed to the general principles of the new IAEA regulations, consisting of a lead container with an external structure giving both mechanical resistance to shock and thermal protection against fire. This was demonstrated with tests on models. The system must have some merit as it has been adopted for the BR2 container.

The escort was requested by the competent authority as a supplementary measure merely because there are no Italian road regulations for this type of shipment.

*** Bogatyrev, V. K., At. Energ. (USSR), *13* (1), 68 (1962).

A. K. GANGULY (India): I understand that casks are tested for integrity in case of fire. Does this mean that there is no risk of the irradiated fuel leaking out if the fuel element melts in a fire?

F. R. MARCUS (Denmark): In order to avoid fuel or canning material melt-down in the case of accidental loss of coolant, the regulations require casks to be designed in such a way that the fission heat can be safely removed.

A. K. GANGULY (India): Is any standard set for the maximum curie activity per kilogramme weight of shipping cask or is there any equivalent specification?

F. R. MARCUS (Denmark): The regulation I have just mentioned may limit the permissible amount of fuel for a given cask. However, no general limit is specified as it would depend on fuel geometry, burn-up, canning material, etc.

Paper P/184 (presented by N. F. C. Bishop)

DISCUSSION

K. MAHMOUD (United Arab Republic): Were there any serious accidents during routine transport either within the United Kingdom or during transport overseas?

N. F. C. BISHOP (United Kingdom): Yes. Details are given in the paper. The most serious accident was a head-on collision between a vehicle carrying the cast-iron Chapelcross container and a lorry loaded with railway wheels and axles. The flask was not damaged.

K. MAHMOUD (United Arab Republic): Do you think that the cost per mile of sea or land transport will be less for longer distances?

N. F. C. BISHOP (United Kingdom): I am not sure. It may be possible to negotiate lower rates per mile for longer journeys. However, lower charges are more usually obtained for *regular* journeys.

J. W. LANGHAAR (United States of America): The paper states that in the case of the MTR flask, British Railways required strengthening of the holding-down arrangements and demonstration by calculation that the wagon frame would absorb some of the energy in an impact accident. Are there specific requirements in these respects? What assumptions were made and what methods were used to calculate the energy of the wagon? What other restrictions do British Railways apply to such shipments?

N. F. C. BISHOP (United Kingdom): There are no special requirements. British Railways accept that the wagon frame can be used as an impact frame. As stated in the paper, the flask will withstand only a 12 ft drop test, and therefore an impact frame is required to satisfy the IAEA regulatory recommendations. The calculations were made by flask and wagon manufacturers. I have no details available, but the principal assumptions were that the wagon is travelling at about 55 mph, that it hits a bridge and that, although the wagon is deformed, the flask does not hit the bridge.

British Railways have no special restrictions—they merely apply the IAEA recommendations.

J. W. LANGHAAR (United States of America): It is noted that there is a trend in the United Kingdom towards solid steel or malleable cast-iron casks rather than lead-filled ones, because better resistance to fire is expected. Has a comparison been made of the economics of the different designs, taking into account both initial cost and shipping cost? Has the degree of fire resistance been specified, so that it may be determined whether or not a particular design meets the requirements?

N. F. C. BISHOP (United Kingdom): As stated in the paper, we have flasks of all types of construction—lead/steel, cast-iron and steel. Recently, most of our experience has been with all-steel flasks but we are not prejudiced against other forms of combination. A capital-cost comparison between lead/steel and allsteel showed virtually no difference in these two types.

We accept the fire test of ½ hour and 810 °C. In the case of large containers, approval is sought on the basis of calculations using specified heat absorption coefficients.

B. G. DUBOVSKY (USSR): Do the standards you have worked out for the transport of irradiated and non-irradiated fuel elements take into account the possibility that the elements may inadvertently fall into water and produce a spontaneous chain reaction?

N. F. C. BISHOP (United Kingdom): Yes, the IAEA regulations take this into account.

Paper P/448 (presented by A. de Streel)

DISCUSSION

K. MAHMOUD (United Arab Republic): How often do you make the inspection necessary for checking the structural integrity of the casks?

A. de STREEL (Belgium): The containers have not yet been used. Tests of structural strength are made during construction: in particular, weld tests and internal pressure tests are carried out. The question of weld-strength tests after fabrication has not yet been considered as far as I know.

Compte rendu de la séance 3.8

Sûreté des usines de traitement chimique; transport du combustible

Président: H. N. Sethna (Inde)

Mémoire P/508 (présenté par J. Tadmor)

DISCUSSION

J. A. McBRIDE (Etats-Unis d'Amérique): L'application des principes décrits dans votre mémoire a-t-elle conduit à des modifications de site, de procédés ou de composants pour certaines installations projetées?

J. TADMOR (Israël): Le guide a été préparé pour l'étude de l'évaluation des risques présentés par le laboratoire chaud qui devait être construit au Centre nucléaire israélien de Nahal Soreq. Les spécialistes de la sûreté savent que ceux qui construisent et utilisent des installations nucléaires n'entreprennent pas avec enthousiasme les études d'évaluation de risques, et nous avons pensé que ce guide pourrait faciliter cette étude.

A. K. GANGULY (Inde): Je voudrais insister sur le fait que les évaluations de risques pour des usines projetées ou en service ont souvent tendance à donner le plus d'importance à la sécurité en fonctionnement normal et en cas d'accident maximal prévisible. Entre ces deux conditions se trouve tout le domaine des accidents possibles, qui se produisent parfois. L'effet cumulatif de tels accidents sur la durée de vie utile d'une usine devrait être pris en considération pour décider de la possibilité de placer l'usine dans un environnement donné. Les exigences de confinement en vue de garantir la sûreté dans le cas de tels accidents peuvent être très rigoureuses.

J. TADMOR (Israël): Je suis entièrement d'accord avec le Dr Ganguly pour estimer qu'en raison de la plus grande probabilité d'accidents relativement mineurs, il faut en faire l'analyse et les évaluer avec autant de soin que pour l'accident maximal prévisible. Ceci est d'ailleurs souligné dans la dernière partie de notre mémoire. Cependant, si des précautions sont prises pour tenir compte de tous les accidents possibles, pour une usine donnée, et si l'évaluation du confinement, des doses de radiation et de la contamination ambiante montre qu'une usine est acceptable sur un certain site eu égard à l'accident maximal prévisible, je pense que l'usine serait aussi acceptable si l'on considère des accidents relativement moins graves et même plus fréquents.

Mémoire P/278 (présenté par J. A. McBride)

DISCUSSION

A. K. GANGULY (Inde): Dispose-t-on de données indiquant la fréquence relative des accidents dus aux rayonnements et des accidents dus à d'autres causes (classiques) dans le fonctionnement d'une telle usine?

J. A. McBRIDE (Etats-Unis d'Amérique): Je n'ai pas de chiffres précis, mais les accidents dus aux rayonnements (c'est-à-dire les expositions supérieures aux limites fixées administrativement) sont moins fréquents que les accidents industriels ordinaires nécessitant des premiers soins ou davantage.

B. G. DUBOVSKY (URSS): Quelle est la méthode de contrôle de la concentration de ^{235}U quand on traite de l'uranium enrichi?

J. A. McBRIDE (Etats-Unis d'Amérique): Administrativement, on peut contrôler la concentration dans les procédés discontinus en contrôlant la quantité de combustible chargé dans l'appareil de dissolution, et par des analyses courantes sur tous les circuits d'uranium dans des récipients sûrs au point de vue de la criticité avant le transfert dans des systèmes non sûrs. Les appareils de dissolution sont inspectés pour déceler des résidus non dissous avant toute nouvelle charge de combustible. La concentration de TBP dans l'extraction est soigneusement contrôlée, et les circuits critiques ont des signaux d'alarme de débit pour annoncer toute interruption de l'écoulement, qui pourrait, autrement, conduire à une augmentation dangereuse de la concentration en uranium. On peut aussi utiliser des détecteurs de neutrons pour signaler que les concentrations en uranium augmentent et tendent vers le point critique. A ce sujet, je vous prie de vous reporter au tableau 2 du mémoire.

B. G. DUBOVSKY (URSS): Mesurez-vous le degré de sous-criticité de l'appareillage chimique quand vous traitez des matériaux fissiles?

J. A. McBRIDE (Etats-Unis d'Amérique): En général, les détecteurs de neutrons utilisés donnent les flux de neutrons en mesurant l'intensité de la réaction (α, n); ils indiquent ainsi la teneur en plutonium des solutions contrôlées. Dans quelques cas, on utilise la multiplication des neutrons, et dans ces cas, les instruments indiquent le degré de sous-criticité du système.

Mémoire P/280 (présenté par M. Levenson)

DISCUSSION

K. D. B. JOHNSON (Royaume-Uni): Si vous aviez à refaire le projet d'une usine semblable, y apporteriez-vous des modifications sur la base de votre expérience?

M. LEVENSON (Etats-Unis d'Amérique): Si nous devions refaire nos plans, nous ferions une cellule sous air ronde, comme la cellule sous argon: la meilleure visibilité, et la possibilité d'utiliser des appareils de manipulation, tels que des treuils et des manipulateurs, sur des rails sans fin constituent un avantage important pour éviter des arrêts dus à des équipements défectueux.

K. D. B. JOHNSON (Royaume-Uni): Si l'installation devait être conçue pour des quantités de combustible 10 ou 100 fois supérieures, la philosophie de votre conception serait-elle très différente?

M. LEVENSON (Etats-Unis d'Amérique): Il est difficile de distinguer, parmi les changements que nous ferions, ceux relatifs à la capacité et ceux dus au passage d'une installation de recherche à une installation de production. La capacité de l'équipement de traitement actuel pourrait naturellement être accrue en travaillant à trois postes sept jours par semaine au lieu du poste unique actuel, cinq jours par semaine. Au point de vue de la sécurité, je ne vois aucun changement à apporter: la quantité de produits de fission et de matériau enrichi déjà traitée est si importante qu'une augmentation de la capacité ne poserait pas de nouveaux problèmes.

B. G. DUBOVSKY (URSS): Quels détecteurs utilisez-vous pour la mesure des neutrons dans des champs γ intenses avec des intensités d'environ 10^6R/h, et quel est le degré de sûreté de leur fonctionnement?

M. LEVENSON (Etats-Unis d'Amérique): Les neutrons ne sont pas mesurés dans l'installation de cycle de combustible de EBR-II, et nous n'avons pas de données sur le fonctionnement des détecteurs dans un tel champ.

Mémoire P/84 (présenté par D. Breton)

DISCUSSION *

J. SCHMETS (Belgique): Les mémoires qui viennent d'être présentés sur les études de criticité montrent l'excellente concordance entre les calculs et les expériences. Peut-on se fier au même degré aux calculs de masse critique pour des matériaux fissiles à l'état solide, tels que des poudres?

D. BRETON (France): Nous n'avons pas d'expérience pratique concernant les matériaux solides et les poudres. Pour les poudres, il faudrait résoudre le problème de la densité.

B. G. DUBOVSKY (URSS): Il serait intéressant de comparer les calculs faits par la méthode des « dimensions équivalentes » ** aux résultats de vos expériences sur l'interaction entre réacteurs sous-critiques.

D. BRETON (France): Nous n'avons pas vu de description de la méthode à laquelle vous avez fait allusion, mais, dès que nous en aurons pris connaissance, nous serons heureux de comparer les résultats.

Mémoire P/381 (présenté par B. G. Dubovsky)

DISCUSSION

M. LEVENSON (Etats-Unis d'Amérique): Je voudrais poser une question à M. Dubovsky concernant le mémoire P/381 et à M. Breton concernant le mémoire P/84. Pensez-vous que nous sommes déjà en mesure de faire confiance aux résultats des calculs pour déterminer les paramètres des installations, par exemple pour de nouvelles installations de traitement, ou croyez-vous qu'il est nécessaire d'avoir un prototype de chaque nouveau type d'installation pour des essais nucléaires avant la construction définitive?

B. G. DUBOVSKY (URSS): Dans le cas des récipients de géométrie régulière (sphères, cylindres, etc.), la dimension de la charge admissible peut être calculée avec une précision suffisante (de l'ordre de 10%), et il est suffisant de faire quelques vérifications pour différentes concentrations d'uranium et de plutonium.

Dans le cas des géométries irrégulières (cylindres à base conique ou hémisphérique, etc.), il faut prendre plus de précautions avant de faire confiance aux calculs: il faut faire des expériences de contrôle pour chaque géométrie de récipient, et comparer les résultats aux calculs de laplaciens. A ce sujet, je voudrais vous signaler un article de V. K. Bogatyrev ***.

D. BRETON (France): La question est assez complexe. Dans le programme du CEA, nous avons fait des expériences sur des éléments de géométrie simple pour vérifier la validité des méthodes et des constantes. Nous avons réservé les calculs par la méthode de Monte-Carlo pour les études relatives aux éléments de géométrie compliquée. Grâce à la rapidité des calculateurs, nous pensons que cette méthode peut être utilisée fréquemment, et conduire à une réduction du nombre des expériences qui devront être réalisées.

Mémoire P/422 (présenté par F. R. Marcus)

DISCUSSION

E. IANSITI (Italie): Votre mémoire déclare que le transport italien d'Ispra à Casaccia ne respectait pas

* Voir aussi la discussion du mémoire P/381.

** Dubovsky, B. G., *et al.*, Atomnaya energia, *16* (1), 16, 1964.

*** Bogatyrev, V. K., Atomnaya Energia, *13* (1), 68, 1962.

les règles approuvées. Ceci n'est pas entièrement exact. Ce transport non seulement était conforme au règlement des Etats-Unis de 10.CFR Part 72, mais de plus, il était prévu pour pouvoir supporter une chute de 10 m suivie d'incendie. Bien que les plans en aient été faits il y a deux ans, le conteneur était conforme aux principes généraux du nouveau règlement de l'AIEA: il comprenait un récipient en plomb et une structure extérieure assurant la résistance mécanique contre les chocs et la protection thermique contre l'incendie. Ceci a été démontré par des essais sur des modèles. Ce système doit présenter des avantages car il a été adopté comme conteneur pour le réacteur BR-2.

L'escorte a été exigée par les autorités compétentes comme mesure supplémentaire simplement parce qu'il n'y a pas de règlement routier en Italie pour ce type de transport.

A. K. GANGULY (Inde): Je pense que l'on essaie les châteaux pour déterminer leur intégrité en cas d'incendie. Ceci veut-il dire qu'il n'y a pas de risque de fuite du combustible irradié si les éléments de combustible fondent au cours d'un incendie?

F. R. MARCUS (Danemark): Pour éviter la fusion du combustible ou du matériau de gainage en cas de perte accidentelle du réfrigérant, les règlements exigent que les châteaux soient conçus de manière que la chaleur de fission puisse être éliminée de façon sûre.

A. K. GANGULY (Inde): Y a-t-il une norme fixée pour l'activité maximale en curies par kilogramme de poids du château de transport, ou y a-t-il un règlement équivalent?

F. R. MARCUS (Danemark): Le règlement que je viens de mentionner peut limiter la quantité de combustible autorisée pour un château donné. Cependant aucune limite générale n'est spécifiée, elle devrait dépendre de la géométrie du combustible, du taux de combustion, du matériau de gainage, etc.

Mémoire P/184 (présenté par N. F. C. Bishop)

DISCUSSION

K. MAHMOUD (République arabe unie): Y a-t-il eu des accidents sérieux pendant des transports courants, soit à l'intérieur du Royaume-Uni, soit au cours des transports au-delà des mers?

N. F. C. BISHOP (Royaume-Uni): Oui. Les détails sont donnés dans le mémoire. L'accident le plus sérieux a été une collision frontale entre un véhicule transportant le conteneur en fonte de Chapelcross et un camion chargé de roues et d'axes de wagons. Le conteneur n'a pas été endommagé.

K. MAHMOUD (République arabe unie): Pensez-vous que le coût au kilomètre des transports maritimes ou terrestres diminuera pour les distances plus grandes?

N. F. C. BISHOP (Royaume-Uni): Je n'en suis pas sûr. Il peut être possible de négocier des tarifs kilométriques plus bas pour de plus grands trajets. Cependant, c'est sur les itinéraires *réguliers* que l'on obtient en général les coûts les plus bas.

J. W. LANGHAAR (Etats-Unis d'Amérique): Le mémoire indique que, dans le cas du château MTR, les Chemins de fer britanniques ont exigé le renforcement du système d'amarrage et la démonstration par le calcul que le châssis du wagon pourrait absorber une partie de l'énergie dans un accident avec impact. Y a-t-il des exigences spécifiques à ce point de vue? Quelles hypothèses ont été faites et quelles méthodes avez-vous utilisées pour calculer l'énergie du wagon? Quelles autres restrictions les Chemins de fer britanniques appliquent-ils à de tels transports?

N. F. C. BISHOP (Royaume-Uni): Il n'y a pas d'exigences spéciales. Les Chemins de fer britanniques acceptent que le châssis du wagon soit utilisé comme châssis d'impact. Comme l'indique notre communication, le récipient ne peut supporter qu'un essai de chute de 12 pieds, et il faut donc un châssis d'impact pour satisfaire au règlement de l'AIEA. Les calculs ont été faits par les constructeurs du récipient et du wagon. Je ne dispose pas de détails, mais les hypothèses principales étaient que le wagon se déplaçait à environ 55 miles par heure, qu'il heurtait un pont et que, bien que le wagon ait été déformé, le récipient ne heurtait pas le pont.

Les Chemins de fer britanniques n'ont pas de restrictions spéciales, ils se contentent d'appliquer les recommandations de l'AIEA.

J. W. LANGHAAR (Etats-Unis d'Amérique): Vous indiquez qu'il y a au Royaume-Uni une certaine tendance à utiliser des châteaux en acier massif ou en fonte malléable plutôt que des châteaux remplis de plomb, dans l'espoir d'obtenir une meilleure résistance à l'incendie. A-t-on fait une comparaison des données économiques pour les différents types, en tenant compte à la fois du coût initial et du coût de transport? Le degré de résistance à l'incendie a-t-il été spécifié, de façon à pouvoir déterminer si un modèle donné satisfait aux conditions exigées?

N. F. C. BISHOP (Royaume-Uni): Comme l'indique le mémoire, nous avons des récipients de tous les types de construction, plomb/acier, fonte et acier. La plupart de nos expériences récentes l'ont été avec des récipients en acier mais nous n'avons pas de parti pris contre d'autres combinaisons. La comparaison des investissements entre les types plomb/acier et tout acier a montré

qu'il n'y a pratiquement pas de différences entre ces deux types.

Nous acceptons l'essai d'incendie d'une demi-heure à 810 °C. Dans le cas des grands récipients, l'autorisation est demandée sur la base de calculs utilisant des coefficients d'absorption de la chaleur spécifiés.

B. G. DUBOVSKY (URSS): Les normes que vous avez établies pour le transport des éléments de combustible irradiés et non irradiés envisagent-elles la possibilité que les éléments tombent accidentellement à l'eau et donnent une réaction en chaîne spontanée?

N. F. C. BISHOP (Royaume-Uni): Oui, les règlements de l'AIEA tiennent compte de cette éventualité.

Mémoire P/448 (présenté par A. de Streel)

DISCUSSION

K. MAHMOUD (République arabe unie): Quelle est la fréquence d'inspection nécessaire pour vérifier l'intégrité de la structure des châteaux?

A. DE STREEL (Belgique): Les châteaux n'ont pas encore été utilisés. Les essais de résistance sont faits pendant la construction: en particulier on vérifie les soudures et on fait des essais sous pression. Le problème des essais de résistance des soudures après fabrication n'a pas, à ma connaissance, été envisagé.

Протокол заседания 3.8

Химическая переработка, транспортировка топлива безопасность реакторов

Председатель: **Х. Н. Сетна (Индия)**

Доклад P/508 (представил Й. Тадмор)

ДИСКУССИЯ

Дж. А. МАК-БРАЙД (США): Внесены ли в результате применения принципа, описанного в докладе, изменения в отношении выбора места расположения реактора, технологических систем или узлов для какой-либо из предложенных установок?

Й. ТАДМОР (Израиль): Подготовлено руководство по изучению и оценке опасностей для горячей лаборатории, которая будет построена в Израильском ядерном центре Нагаль-Сорек. Специалисты по обеспечению безопасности реакторов убедились, что строители и операторы ядерных установок неохотно проводят работы по оценке опасностей, и мы считали, что это руководство ускорит проведение таких работ.

А. К. ГАНГУЛИ (Индия): Я хотел бы отметить, что оценка опасностей проектируемых или действующих установок часто направлена на обеспечение безопасности при нормальной работе или в случае максимально возможной аварии. Между этими двумя крайними случаями лежит целая область возможных аварий, которые иногда случаются. Общий эффект таких аварий за весь срок службы установки необходимо рассмотреть с точки зрения приемлемости этой установки для данной окружающей среды. Для обеспечения безопасности в случае таких аварий требования к противоаварийным оболочкам должны быть очень жесткими.

Й. ТАДМОР (Израиль): Я полностью согласен с д-ром Гангули в том, что ввиду более высокой вероятности относительно небольших аварий их необходимо анализировать и оценивать так же внимательно, как и максимально возможные аварии. Фактически это положение подчеркивается в последней части нашего доклада. Однако если для данной установки предусмотрены меры предосторожности для возможных аварий и если оценка степени удержания радиоактивных продуктов, дозы облучения и загрязнения окружающей среды показывает, что установка приемлема для определенного участка с точки зрения максимально возможной аварии, то, по моему мнению, она также будет приемлема и с точки зрения относительно меньших и даже более частых аварий.

Доклад P/278 (представил Дж. А. Мак-Брайд)

ДИСКУССИЯ

А. К. ГАНГУЛИ (Индия): Не можете ли Вы привести цифровые данные, показывающие относительное количество радиационных пораже-

ний и несчастных случаев, вызванных другими (обычными) причинами, за время эксплуатации такой установки?

Дж. А. МАК-БРАЙД (США): У меня нет под рукой точных цифровых данных, но количество радиационных поражений (то есть облучение свыше установленных норм) меньше количества обычных промышленных несчастных случаев, для которых требуется скорая помощь или более серьезное медицинское вмешательство.

Б. Г. ДУБОВСКИЙ (СССР): Как осуществляется контроль концентрации U^{235} при переработке обогащенного урана?

Дж. А. МАК-БРАЙД (США): Контроль концентрации в периодических процессах может осуществляться путем контроля количества топлива, загружаемого в аппарат для растворения, а также путем серийных анализов всех урановых потоков в сосудах, безопасных с точки зрения образования критической массы, перед их подачей в небезопасные системы. Перед перегрузкой топлива производится осмотр аппаратов для растворения, с тем чтобы убедиться в отсутствии нерастворившихся хвостов. В процессе экстракции производится тщательный контроль концентрации трибутилфосфата, а все критические потоки снабжены приборами, которые сигнализируют в случае прекращения потока, что в противном случае может привести к опасному повышению концентрации урана. Можно также использовать нейтронные регистраторы для сигнализации в случае повышения концентрации урана к критической точке. В этой связи я рекомендовал бы вам рассмотреть табл. 2, приведенную в докладе.

Б. Г. ДУБОВСКИЙ (СССР): Измеряете ли Вы степень подкритичности химической аппаратуры при переработке делящегося материала?

Дж. А. МАК-БРАЙД (США): Большей частью используемые регистраторы обнаруживают нейтронный поток путем измерения интенсивности (α, n)-реакции; таким образом, они показывают концентрацию плутония в контролируемых потоках. В нескольких случаях используется размножение нейтронов, и в этих случаях приборы будут показывать степень подкритичности системы.

Доклад P/280 (представил М. Левенсон)

ДИСКУССИЯ

К. Д. Б. ДЖОНСОН (Соединенное Королевство): Если бы Вам снова пришлось проектировать аналогичную установку, то внесли бы в нее какие-либо изменения, исходя из Вашего опыта?

М. ЛЕВЕНСОН (США): Если бы мы снова начали такое проектирование, то мы бы сделали воздушную камеру круглой, подобно аргоновой камере: увеличение обзора и возможность управления оборудованием на гусеничном ходу для перемещения и транспортировки материалов, например, кранами и манипуляторами, является большим преимуществом с точки зрения предотвращения остановок реактора из-за выхода оборудования из строя.

К. Д. Б. ДЖОНСОН (Соединенное Королевство): Если бы пришлось проектировать оборудование на 10- или 100-кратное количество топлива, то насколько серьезно изменился бы Ваш проект?

М. ЛЕВЕНСОН (США): Очень трудно подразделить изменения, которые мы бы внесли в конструкцию, на те, которые относятся к мощности установки, и те, которые связаны с переходом от исследовательской установки к промышленной. Можно, конечно, было бы повысить производительность существующего технологического оборудования введением трехсменной работы и семидневной рабочей недели вместо односменной работы пять дней в неделю в настоящее время. С точки зрения безопасности я не знаю, что нам следовало бы менять: количество продуктов деления и обогащенного материала, которые подвергаются обработке, уже сейчас настолько велико, что увеличение мощности установки не приведет к возникновению новых проблем.

Б. Г. ДУБОВСКИЙ (СССР): Какие чувствительные элементы применяются для регистрации нейтронов в сильных гамма-полях интенсивностью около 10^6 $р/ч$ и насколько они надежны в эксплуатации?

М. ЛЕВЕНСОН (США): Нейтроны не измеряются на установке топливного цикла реактора EBR-II, и у нас нет данных об их эксплуатации в таком поле.

Доклад P/84 (представил Д. Бретон)

ДИСКУССИЯ*

Ж. ШМЕТЦ (Бельгия): В докладах, которые мы только что слышали по критическим исследованиям, показано отличное соответствие между расчетом и экспериментом. Можно ли в равной мере полагаться на расчеты критиче-

* См. также дискуссию к докладу P/381.

ской массы делящихся материалов в твердом состоянии, например порошков?

Д. БРЕТОН (Франция): У нас нет практического опыта по критическим массам твердых материалов и порошков. В отношении порошков необходимо решить проблему плотности.

Б. Г. ДУБОВСКИЙ (СССР): Интересно сравнить расчеты методом «эквивалентных размеров» ** с результатами Ваших экспериментов по взаимодействию подкритических реакторов.

Д. БРЕТОН (Франция): Мы не видели описания метода, о котором Вы говорите, но, как только мы ознакомимся с ним, мы будем рады сравнить результаты.

Доклад P/381 (представил Б. Г. Дубовский)

ДИСКУССИЯ

М. ЛЕВЕНСОН (США): У меня есть вопрос к г-ну Дубовскому по докладу P/381 и к г-ну Бретону по докладу P/84. Думаете ли Вы, что мы можем полагаться на результаты расчета для определения проектных параметров, например для новых технологических установок, или Вы считаете, что необходимо смоделировать любой предлагаемый новый тип установки для проведения ядерных испытаний еще до строительства этой установки?

Б. Г. ДУБОВСКИЙ (СССР): В случае сосудов правильной геометрии (сферы, цилиндры и т. д.) размер допустимой загрузки может быть рассчитан с достаточной точностью (около 10%). При этом достаточно проводить местные измерения различных концентраций урана и плутония.

Если же речь идет о сосудах неправильной геометрии (цилиндры с коническим дном или полусферическим дном и т. д.), тогда к расчетам следует подходить с большей осторожностью: необходимо поставить контрольные эксперименты для каждого класса геометрии и результаты сравнить с расчетами кривизны. В этой связи я хотел бы, чтобы Вы обратились к статье В. К. Богатырева ***.

Д. БРЕТОН (Франция): Это довольно сложный вопрос. По программе Комиссариата по атомной энергии Франции мы провели эксперименты с элементами простой геометрии, чтобы проверить справедливость методов и констант. Мы использовали расчеты методами типа Монте-Карло для изучения элементов сложной геометрии. Принимая во внимание скорость работы вычислительных машин, мы считаем, что этот метод может быть часто использован, в результате чего будет сокращено число экспериментов, которые необходимо поставить.

Доклад P/422 (представил Ф. Р. Маркус)

ДИСКУССИЯ

Э. ИАНСИТ (Италия): В Вашем докладе утверждается, что перевозка из Испры в Казачча не отвечала утвержденным стандартам. Это не совсем точно. Эта перевозка производилась не только в соответствии с проектом соответствующей инструкции, подготовленным в США (10.CFR, часть 72), но была рассчитана таким образом, чтобы выдержать падение с высоты 9,2 *м* с последующим пожаром. Хотя контейнер был спроектирован два года тому назад, он соответствует общим принципам новых правил МАГАТЭ. Этот свинцовый контейнер снабжен внешними конструкционными элементами для обеспечения механической прочности против ударов и тепловой защитой против огня. Это было проверено путем испытаний на моделях. Эта система должна обладать некоторыми достоинствами, поскольку она была принята для контейнера BR-2.

Поскольку в Италии не разработаны дорожные правила для перевозок такого типа, компетентные власти обеспечили эскортирование в качестве дополнительной меры.

А. К. ГАНГУЛИ (Индия): Как я понимаю, контейнеры испытываются на сохранение целостности в случае пожара. Означает ли это, что нет никакой опасности утечки облученного топлива, если тепловыделяющий элемент расплавится в случае возникновения пожара?

Ф. Р. МАРКУС (Дания): Для того чтобы избежать расплавления топлива или материала оболочки в случае аварийной потери теплоносителя, правила предписывают, чтобы конструкция контейнера позволяла безопасно отводить тепло, выделяющееся при распаде продуктов деления.

А. К. ГАНГУЛИ (Индия): Существуют ли какие-нибудь стандарты, определяющие максимальную активность в кюри, приходящуюся на 1 *кг* веса транспортного контейнера, или имеется какая-либо эквивалентная спецификация?

П. Р. МАРКУС (Дания): Инструкция, которую я только что упомянул, может ограничивать допустимое количество топлива для данного контейнера. Однако не установлено никакого общего предела, поскольку он будет зависеть от геометрии топлива, его выгорания, материала оболочки и т. д.

** Б. Г. Дубовский и др. Атомная энергия, *16*, 1, 16 (1964).

*** В. К. Богатырев. Атомная энергия, *13*, 1, 68 (1962).

Доклад P/184 (представил Н. Ф. К. Бишоп)

ДИСКУССИЯ

К. А. МАХМУД (ОАР): Были ли случаи серьезных аварий во время обычной перевозки в Великобритании или за ее пределами?

Н. Ф. К. БИШОП (Соединенное Королевство): Я не уверен в этом. Может представиться возможность начать переговоры о понижении тарифов за 1 *км* пути при перевозке на более длинные расстояния. Однако более низкие тарифы обычно применяют в случае регулярных перевозок.

К. А. МАХМУД (ОАР): Думаете ли Вы, что при перевозке на большие расстояния наземным или морским транспортом стоимость перевозки, приходящаяся на 1 *км* пути, будет меньше?

Н. Ф. К. БИШОП (Соединенное Королевство): Да, подробности приведены в докладе. Самой серьезной аварией было прямое столкновение между автомобилем, на котором был установлен контейнер из литого чугуна из Чэпел-Кросса, и грузовым автомобилем с грузом железнодорожных колес и осей. Контейнер при этом не получил повреждений.

Дж. У. ЛЭНГХАР (США): В докладе указывается, что в случае контейнеров для тепловыделяющих элементов типа MTR Управление британских железных дорог требует повысить прочность креплений, удерживающих контейнер в вагоне, а также провести демонстрационный расчет, который бы показал, что рама вагона поглотит часть энергии в случае столкновения. Существуют ли особые требования в этом отношении? Какие допущения были сделаны и какие методы использовались для расчета энергии вагона? Какие другие ограничения выдвигаются Управлением британских железных дорог к таким перевозкам?

Н. Ф. К. БИШОП (Соединенное Королевство): Нет никаких специальных требований. Управление британских железных дорог согласилось с тем, что рама вагона может служить для восприятия удара при столкновении. Как указывалось в докладе, контейнер выдерживает только испытание на падение с высоты 3,6 *м*. Поэтому рама вагона, воспринимающая удар, должна удовлетворять нормативным рекомендациям МАГАТЭ. Эти расчеты были проведены фирмами — изготовителями контейнера и вагона. Я не знаю потребностей, но основное допущение заключалось в том, что при столкновении вагона, движущегося со скростью примерно 88 *км/ч*, с мостом вагон деформируется, но контейнер не сталкивается с мостом.

Управление британских железных дорог не имеет специальных ограничений, оно просто использует рекомендации МАГАТЭ.

Дж. У. ЛЭНГХАР (США): Отмечено, что в Великобритании существует тенденция к предпочтительному использованию контейнеров из твердой стали или из ковкого чугуна, а не контейнеров со свинцовым заполнением, поскольку первые обладают большой стойкостью к действию огня. Проводилось ли экономическое сравнение различных конструкций контейнеров с учетом первоначальной стоимости контейнера и стоимости перевозки? Установлена ли степень огнестойкости контейнера, которая бы позволила определить соответствие данной конструкции этим требованиям?

Н. Ф. К. БИШОП (Соединенное Королевство): Как указывалось в докладе, у нас имеются контейнеры всех типов конструкций — из свинца и стали, литого чугуна и стали. За последнее время бóльший опыт приобретен нами с контейнерами, изготовленными целиком из стали, но у нас нет предубеждений против других сочетаний. Сравнение капитальных затрат между контейнерами из свинца и стали и контейнерами, полностью изготовленными из стали, практически не показало никакой разницы между этими двумя типами.

Испытание на огнестойкость мы проводим при температуре 810° С в течение 30 *мин*. В случае больших контейнеров оценку проводим расчетным путем, используя определенные коэффициенты теплопоглощения.

Б. Г. ДУБОВСКИЙ (СССР): Предусматривают ли стандарты, которые Вы разработали для перевозки облученных и необлученных тепловыделяющих элементов, возможность аварийного попадания этих элементов в воду и начала спонтанной цепной реакции?

Н. Ф. К. БИШОП (Соединенное Королевство): Да, правила МАГАТЭ учитывают возможность таких случаев.

Доклад P/448 (представил А. де Стрит)

ДИСКУССИЯ

К. А. МАХМУД (ОАР): Как часто Вы проводите осмотр, необходимый для проверки конструкционной прочности контейнеров?

А. ДЕ СТРИТ (Бельгия): Эти контейнеры еще не применялись. Испытания на прочность конструкции проводятся при изготовлении контейнера, в частности особое внимание уделяется испытанию сварных швов и внутреннего давления. Насколько мне известно, вопрос об испытаниях прочности сварных швов после изготовления контейнеров еще не рассматривался.

Acta de la Sesión 3.8

Seguridad en las instalaciones de tratamiento químico; transporte del combustible

Presidente: H. N. Sethna (India)

Documento /508 (presentado por J. Tadmor)

DISCUSIÓN

J. A. McBRIDE (Estados Unidos de América): ¿La aplicación de los principios descritos en la memoria ha producido modificaciones con respecto al lugar de emplazamiento, métodos de proceso, o componentes de cualquier instalación propuesta?

J. TADMOR (Israel): Las normas fueron preparadas para llevar a cabo un estudio de la evaluación de riesgos del laboratorio caliente que va a ser construído en el Centro Nuclear Israelí de Nahal Soreq. El personal de seguridad se da cuenta de que los constructores y operadores de fábricas nucleares no llevan a cabo voluntariamente la evaluación de riesgos y nosotros pensamos que estas normas facilitarían un estudio de este tipo.

A. K. GANGULY (India): Me gustaría comentar que la evaluación de riesgos en fábricas en diseño o en operación tiende con frecuencia a aumentar la seguridad durante la operación normal y en relación con un máximo accidente posible. Entre estas dos condiciones existe todo un intervalo de accidentes probables, que suceden ocasionalmente. El efecto acumulativo de tales accidentes durante el tiempo total de operación de una fábrica debería ser estudiado desde el punto de vista de la aceptación de la construcción de la fábrica en un lugar determinado. Los requesitos de confinamiento tal vez tengan que ser bastante exigentes para garantizar seguridad en tales accidentes.

J. TADMOR (Israel): Estoy completamente de acuerdo con el Dr. Ganguly de que, en vista de que hay una mayor probabilidad de accidentes relativamente pequeños, deberían ser analizados y evaluados tan cuidadosamente como el máximo accidente probable (MAP). En efecto, esto está recalcado en la última parte de nuestra memoria. Sin embargo, si las seguridades se dan para enfrentarse con todo el intervalo de accidentes para una fábrica dada, y si la evaluación del confinamiento, dosis de radiación y contaminación de los alrededores muestran que una instalación es aceptable en un cierto sitio desde el punto de vista del MAP, yo creo que también sería aceptable desde el punto de vista de accidentes relativamente más pequeños y más frecuentes.

Documento P/278 (presentado por J. A. McBride)

DISCUSIÓN

A. K. GANGULY (India): ¿Hay datos disponibles que indiquen la relativa frecuencia de daños debidos a radiación y a otras causas (convencionales) en la operación de tal instalación?

J. A. McBRIDE (Estados Unidos de América): No tengo datos especificos a mano, pero los accidentes de radiación (por ejemplo, exposición por encima de los límites establecidos administrativamente) son menos frecuentes que aquellos accidentes industriales ordinarios que requieren tratamiento de primeros auxilios o más graves.

B. G. DUBOVSKY (URSS): ¿Cuál es el método para controlar la concentración de ^{235}U en procesos en que se maneja uranio enriquecido?

J. A. McBRIDE (Estados Unidos de América): Administrativamente, la concentración en procesos por cargas puede ser controlada por medida de la cantidad de combustible cargado al disolvedor, y por análisis rutinario de todas las corrientes de uranio en los depósitos críticamente seguros antes de ser trasvasadas a sistemas no seguros. Los disolvedores son inspeccionados para controlar la cantidad de material no disuelto antes de llevar a cabo una nueva carga de combustible. La concentración de fosfato de tributilo en la sección de extracción con disolventes es cuidadosamente controlada, y en las corrientes críticas hay alarmas para indicar cualquier interrupción del flujo, que podría de otro modo conducir a un aumento en la concentración de uranio por encima del límite de seguridad. Se pueden emplear también monitores de neutrones para indicar que las concentraciones de uranio están aumentando hacia el punto crítico. Respecto a este punto le remito a la Tabla 2 de la memoria.

B. G. DUBOVSKY (URSS): ¿Miden Vds. el grado de subcriticidad de los aparatos químicos que se emplean en los procesos de material fisible?

J. A. McBRIDE (Estados Unidos de América): En su mayoría, los monitores de neutrones en uso detectan el flujo de neutrones midiendo la intensidad de la reacción (α, n); por consiguiente indican la concentración de plutonio en las corrientes que son controladas.

En unos pocas casos se usa multiplicación de neutrones, y entonces los instrumentos indicarán el grado de subcriticidad del sistema.

Documento P/280 (presentado por M. Levenson)

DISCUSIÓN

K. D. B. JOHNSON (Reino Unido): ¿Si Vd. tuviera que empezar otra vez el diseño de una fábrica semejante, basandose en su experiencia, haría Vd. cambios?

M. LEVENSON (Estados Unidos de América): Si tuvieramos que empezar otra vez el diseño haríamos la celda redonda como la celda de argón; el aumento de la visibilidad y la oportunidad de operar parte del equipo, como grúas y manipuladores, sobre carriles sin fin, es una gran ventaja desde el punto de vista de evitar paradas debido a fallos del equipo.

K. D. B. JOHNSON (Reino Unido): Si la instalación tuviera que ser diseñada para una cantidad de combustible 10 ó 100 veces mayor, ¿su criterio de diseño sería muy diferente?

M. LEVENSON (Estados Unidos de América): Es difícil distinguir entre los cambios que haríamos relativos a la capacidad de la instalación con aquellos relativos al paso de una fábrica de investigación a una de producción. La capacidad del equipo podría ser aumentada trabajando siete días por semana a tres turnos diarios en vez de un solo turno que trabaja cinco días por semana, como sucede actualmente. Desde el punto de vista de la seguridad, nada cambiaría: las cantidades de productos de fisión y de material enriquecido que están siendo tratadas son ya tan grandes que un aumento de la capacidad no plantearía nuevos problemas.

B. G. DUBOVSKY (URSS): ¿Qué elementos sensibles se emplean para registrar el flujo de neutrones en presencia de fuerte actividad gamma y con intensidades de 10^6 r/h, y cómo es su funcionamiento?

M. LEVENSON (Estados Unidos de América): El flujo de neutrones no es medido en la instalación del ciclo del combustible del EBR-II y no tenemos datos de su comportamiento en semejante campo.

Documento P/84 (presentado por D. Breton)

DISCUSIÓN *

J. SCHMETS (Bélgica): Las memorias que nos han sido presentadas sobre estudios de criticidad muestran excelente concordancia entre los datos calculados y los experimentales. ¿Se puede confiar de la misma forma en el cálculo de la masa crítica para materiales fisibles en estado sólido, tales como polvos?

D. BRETON (Francia): No tenemos experiencia práctica sobre las masas críticas de materiales sólidos y polvos. Con polvos, hay que resolver el problema de la densidad.

B. G. DUBOVSKY (URSS): Sería interesante comparar los datos calculados por el método de las « dimensiones equivalentes » ** con los resultados de sus experimentos sobre las interacciones de los reactores subcríticos.

D. BRETON (Francia): No hemos visto una descripción del método a que Vd. se refiere, pero tan pronto como estemos familiarizados con él compararemos dichos resultados.

Documento P/381 (presentado por G. B. Dubovsky)

DISCUSIÓN

M. LEVENSON (Estados Unidos de América): Quisiera hacer una pregunta al Sr. Dubovsky sobre su documento P/381 y al Sr. Breton sobre su documento P/84. ¿Creen Vds. que aún podemos confiar en los resultados del cálculo para la determinación de los parametros de diseño, por ejemplo, para nuevas instalaciones, o Vds. creen que es necesario hacer un modelo para cualquier instalación propuesta de nuevo tipo, antes de su construcción?

B. G. DUBOVSKY (URSS): En el caso de depósitos de geometría regular (esferas, cilindros, etc.) la cantidad de carga permisible puede ser calculada con bastante seguridad (alrededor del 10%), y es suficiente para llevar a cabo comprobaciones sobre el terreno para diferentes concentraciones de uranio o plutonio.

En el caso de geometrías irregulares (cilindros con bases cónicas o semiesféricas, etc.) se deben tomar más precauciones respecto a los cálculos; se deberían hacer experimentos de control para cada geometría de los depósitos, y los resultados deberían ser comparados con los cálculos de protección.

Respecto a esto, le remito a Vd. a un artículo de V. K. Bogatyrev ***.

D. BRETON (Francia): La pregunta es más bien complicada. En el programa del CEA hemos llevado a cabo ensayos con elementos de geometría simple para comprobar la validez de los métodos y constantes. Reservamos los cálculos del tipo Monte Carlo para estudios relativos a elementos de geometría complicada. En vista de la velocidad de las máquinas calculadores, creemos que este método puede ser usado frecuentemente, con una considerable reducción en el número de experimentos que se deben hacer.

* Véase también la discusión del documento P/381.

** Dubovsky, B. C., *et al.*, At. Energ. (URSS), 16 (1), 16 (1964).

*** Bogatyrev, V. K., At. Energ. (URSS), *13* (1), 68 (1962).

Documento P/422 (presentado por F. R. Marcus)

DISCUSIÓN

E. IANSITI (Italia): Su memoria establece que el envío italiano de Ispra a Casaccia no satisfacía a las normas aprobadas. Esto no es exacto. El cargamento no solamente estaba de acuerdo con el esquema del reglamento, estadounidense 10.CFR Part 72, sino que también fue concebido de forma que pudiera resistir una caída desde 30 pies, seguida de fuego. Aunque diseñado hace dos años, el recipiente estaba de acuerdo con los principios generales de la nueva reglamentación del OIEA y consitía en un recipiente de plomo con una estructura interna, reuniendo ambos resistencia mecánica contra el choque y protección térmica contra el fuego. Esto se demostró en ensayos realizados con modelos. El sistema debe tener algún mérito porque ha sido adoptado para el recipiente de transporte del BR2.

La escolta fue pedida por la autoridad competente solamente como una medida suplementaria, porque en Italia no hay un reglamento de carreteras para este tipo de envíos.

A. K. GANGULY (India): Tengo entendido que los recipientes de transporte son probados para comprobar su integridad en caso de fuego. ¿Significa esto que no hay peligro de que el combustible irradiado escape al exterior si el elemento combustible se funde con el fuego?

F. R. MARCUS (Dinamarca): Para evitar que el combustible o el material de envainado fundan en el caso de una pérdida accidental de refrigerante, las normas requieren que las vasijas de transporte sean diseñadas de tal forma que el calor de la fisión se pueda eliminar fácilmente.

A. K. GANGULY (India): ¿Hay alguna norma establecida para la máxima actividad en curios por kilogramo de recipiente de transporte, o hay alguna especificación equivalente?

F. R. MARCUS (Dinamarca): La norma que he mencionado puede limitar la cantidad permisible de combustible para un recipiente dado. Sin embargo, no hay especificado ningún límite general y ello dependería de la geometría del combustible, grado de quemado, material de envainado, etc.

Documento P/184 (presentado por N. F. C. Bishop)

DISCUSIÓN

K. MAHMOUD (República Arabe Unida): ¿Hubo accidentes serios durante el transporte rutinario, bien dentro del Reino Unido, bien al extranjero?

N. F. C. BISHOP (Reino Unido): Sí. Se dan detalles en la memoria. El accidente más serio fue una colisión de frente entre un vehículo que transportabla el recipiente de hierro fundido de Chapelcross y un camión cargado con ruedas y ejes de ferrocarril. La vasija no sufrió roturas.

K. MAHMOUD (República Arabe Unida): ¿Dice Vd. que el coste de transporte por milla, marina o terrestre, será menor para distancias más largas?

N. F. C. BISHOP (Reino Unido): No estoy seguro. Puede ser posible lograr más bajos costes por milla para viajes más largos. Sin embargo, normalmente se consiguen costes más bajos para viajes regulares.

J. W. LANGHAAR (Estados Unidos de América): La memoria establece que, en el caso del recipiente para el MTR, los Ferrocarriles Británicos exigieron reforzamiento de los dispositivos de sujeción y demostración mediante cálculos de que la estructura del vagón absorbería parte de la energía en un accidente por impacto. ¿Hay exigencias específicas a este respecto? ¿Qué hipótesis se hicieron y qué métodos se usaron para calcular la energía del vagón? ¿Qué otras restricciones aplican los Ferrocarriles Británicos a tales envíos?

N. F. C. BISHOP (Reino Unido): No hay requisitos especiales. Los Ferrocarriles Británicos admiten que la estructura del vagón puede ser empleada como estructura de impacto. Como se indica en la memoria, el recipiente de transporte soportará solamente una prueba de caída de 12 pies y por tanto se require una estructura de impacto para satisfacer las normas del OIEA. Los cálculos fueron hechos por los constructores del vagón y del recipiente. No tengo detalles a mano, pero las principales hipótesis fueron que el vagón marcha alrededor de 55 millas por hora, que choca contra un puente y que, aunque el vagón sea deformado, el recipiente de transporte no choca contra el puente.

Los Ferrocarriles Británicos no tienen restricciones especiales, meramente aplican las normas del OIEA.

J. W. LANGHAAR (Estados Unidos de América): Se señala que en el Reino Unido hay una tendencia hacia los recipientes de acero o fundición maleable mayor que hacia los rellenados con plomo, porque se espera que tengan una resistencia mejor al fuego. ¿Se ha hecho una comparación económica para los diferentes diseños, teniendo en cuenta tanto el coste inicial como el coste de embarque? ¿Se ha especificado el grado de resistencia al fuego, de forma que se pueda determinar si un cierto diseño reúne o no las condiciones requeridas?

N. F. C. BISHOP (Reino Unido): Como se indica en la memoria, tenemos de todos los tipos de construcción: plomo/acero, fundición y acero. Recientemente, la mayor parte de nuestras experiencias se hicieron con recipientes totalmente de acero; pero no tenemos prejuicios contra otras formas de combinación. La

comparación del coste entre recipientes de plomo/acero y sólo de acero indicó que prácticamente no hay diferencia entre estos dos tipos.

Aceptamos ½ hora y 810 °C para el ensayo de fuego. En el caso de recipientes grandes, la aprobación se busca tomando como base de cálculo coefficientes de absorción de calor especificados.

B. G. DUBOVSKY (URSS): Las normas que Vd. ha empleado ¿tienen en cuenta, para el transporte de elementos combustibles irradiados y no irradiados, la posibilidad de que los elementos combustibles puedan caer inadvertidamente dentro del agua y se pueda producir una reacción en cadena espontánea?

N. F. C. BISHOP (Reino Unido): Sí, las normas del OIEA tienen esto en cuenta.

Documento P/448 (presentado por A. de Streel)

DISCUSIÓN

K. MAHMOUD (República Arabe Unida): ¿Con qué frecuencia hacen Vds. la inspección necesaria para comprobar la integridad estructural del recipiente de transporte?

A. DE STREEL (Bélgica): Los recipientes no se han usado todavía. Los ensayos de resistencia estructural se hacen durante la construcción: particularmente se hacen ensayos de la soldadura y pruebas de presión interna. Que yo sepa, la cuestión sobre las pruebas de resistencia de la soldadura después de la fabricación no ha sido aún considerada.

Session 3.6

REACTOR SAFETY; LOCATION AND CONTAINMENT; REACTOR TESTS

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Reactor safety practice and experience in Canada

By G. C. Laurence, F. C. Boyd, J. H. Jennekens, J. B. Sutherland and P. E. Hamel *

In Canada, the Atomic Energy Control Board is the regulatory agency for atomic energy [1]. It discharges its responsibilities for reactor safety with the assistance and advice of the national and provincial Departments of Health and of Labour and of a Reactor Safety Advisory Committee (RSAC).

The RSAC was formed in 1956; it consisted of scientists and engineers in federal and provincial government departments and other organizations. The RSAC, assisted by the staff of the Atomic Energy Control Board, follows the design and construction of a new reactor and its operation after it is completed. Authorization to start construction is not granted by the Board unless the RSAC believe that a reactor of the proposed type can be built and operated safely on the proposed site. Licence to operate is not granted until the reactor has been constructed and commissioned to the satisfaction of the RSAC and they believe that it will be operated safely.

The federal government's research and development organization, Atomic Energy of Canada Limited (AECL), has its own Review Committee on Reactor Safety to study reactors that are wholly owned and operated by that company. The philosophy of both committees has been influenced by experience with the AECL research reactors NRX and NRU.

The NRX reactor was built when there was little experience to guide the design. The original safety and protective features were elaborate and complicated. The accident in 1952 [2, 3] showed that simpler safety measures with good maintenance would provide better protection than equipment that was too elaborate to keep in good repair. More sceptical appraisal of safety measures has led during the intervening years to important changes in their choice and design. The quantitative assessment of the effectiveness and the reliability of safety measures has been emphasized.

Meanwhile, efforts have been made to set safety standards to guide the designers. Basically, it is felt that risk to the public from the operation of a nuclear reactor should be no greater than from the operation of other kinds of industrial plants of equal economic importance [4, 5]. Safety standards can be expressed in terms of expected frequency of occurrence of disasters and the number of deaths and injuries that they produce.

Such standards are difficult to use. It would take too long to discover from actual experience of accidents whether or not they are being met. Therefore, safety requirements are being expressed in terms of the frequency of dangerous failures of equipment and the effectiveness and the reliability of different divisions of the reactor plant and its equipment [6].

In the discussion that follows, the reactor plant is considered in three divisions: the process equipment, the protective devices and the containment provisions. By considering the design in these divisions separately, it is possible to make realistic estimates of the contribution of each division to the probability of dangerous escape of fission products.

(*a*) The *process equipment* includes all that is necessary for the functioning of a reactor except the protective devices and the containment provisions.

(*b*) The *protective devices* include the automatic scram or shut-down equipment, the emergency cooling system, the emergency power supplies and other devices that are designed to prevent any failure in the process equipment or any operating error from causing damage to the fuel that would permit the release of radioactive fission products.

(*c*) The *containment provisions* include any containment shell or structure, any vapour-suppression provisions and any other equipment that contributes in any way to confining the radioactive material that might accidentally escape from the process equipment.

It is an important principle of safe design that these divisions—process equipment, protective devices and containment provisions—be structurally and operationally independent. In particular, the automatic reactivity regulating system, regarded as part of the process equipment, should be electrically and mechanically very independent of the protective devices so that a failure of one does not incapacitate the other.

Also, no significant radiation injury should be caused outside the plant by a process-equipment failure unless both the protective and containment systems also fail to function.

* Atomic Energy Control Board, Ottawa, Ontario.

Faults that are produced in different components, devices or parts of a system by the same cause are called *cross-linked faults*. Faults that do not result from one common cause are called *independent faults*.

It is difficult to assess the probable rate of cross-linked faults because they are infrequent. It is assumed that by taking great pains to make the three divisions of the plant independent, the probable rate of cross-linked faults [6] can be reduced to less than the probable rate of coincidence of independent faults. The probable rate of independent faults is determined by observation of their occurrence in operation.

The *unreliability* of a protective device, a containment provision, or any component of them, is the fraction of time that it is unable to perform its intended function.

As standards of performance, the following limits have been proposed [6]: the frequency of dangerous faults in the process equipment should not exceed one per year; the unreliability of the protective devices should not exceed 0.003; the unreliability of the containment provisions should not exceed 0.003.

These limits are large enough that they permit some faults during a few years, and knowledge of the frequencies and unreliabilities can be obtained in actual operation in a year or two. On the other hand, these limits are so small that the probability that the three divisions will fail simultaneously from independent faults, and hence lead to injury outside the exclusion area, is acceptably small.

Standards of this kind not only guide the designer but also assist the operator in assessing the behaviour of a plant after it is put into operation. Deterioration in performance must be detected and corrected and the reliability of the equipment must be maintained at the high level that the designer had intended.

PROCESS EQUIPMENT

Reactivity regulation

There has been a continuous evolution of the philosophy and design of the control system of the large reactors [7, 8], especially since the NRX accident [2, 3]. The original NRX regulating system used four regulating rods which were independently controlled by the operator. Reactivity could also be controlled by changing the level of the heavy-water moderator in the reactor vessel. At an early stage, three regulating rods were removed and the remaining one arranged so that it could be automatically controlled when the reactor was at a steady power.

In 1958, the control system was made fully automatic, using moderator-level adjustment as the controlling mechanism [9]. Owing to the large photoneutron source in heavy-water reactors, the entire range from shut-down to full power can be covered without moving or changing ion chambers. Therefore, fully automatic control of start-up as well as continuing operation is feasible with a relatively simple system. Moderator-level adjustment is used as the basic control of the NPD reactor [11] and for the normal automatic start-up of the CANDU reactor in the Douglas Point plant [12].

The most significant development in the control system since 1952 is in the use of the principle of redundancy through triplicated circuits. It was first applied to the protective circuits to reduce the number of shut-downs caused by false signals. When it was found that triplication not only reduced the number of spurious shut-downs but also improved reliability and safety, it was adopted also in the regulating system.

Two advantages accrue from triplication. Each circuit can be tested, adjusted or repaired without interrupting the operation of the reactor or depriving it of a continuous safe control. Also, the performance of one circuit can be checked by the performance of others so that disagreement is revealed and the faulty channel may be rejected from control.

It has been possible to limit physically the maximum rate of addition of reactivity, $(dk/dt)/k$, to about 0.0003 per second without great inconvenience for operation. If a failure of the regulating system occurred, this provision would limit the power that was reached before the protective system became effective.

As much as possible the fail-safe principle has been applied in the control systems: common types of failure will produce a safer condition. A further principle is now being applied: all the components of the control system are used under much less severe conditions than the manufacturers' ratings. Previously, such underratings had been common only in electronic components.

In large reactors, variations of xenon poisoning [10] necessitate control of the reactivity of different parts of the core. Eight booster or enriched rods and four absorber rods were added to CANDU at Douglas Point [12] for this purpose. The movement of these rods will be controlled by a digital computer which will analyse the temperature and flow information from each reactor channel.

With automatic control, the role of the operator is changed. It now becomes one of observing and checking the performance of the equipment to discover any increase in unreliability and of ensuring that faults are not neglected. Provided the operator realises his new role and conscientiously carries out his duties, automatic regulation is safer than manual regulation because his attention is not fully absorbed by a function that is better performed by electrical devices.

Primary cooling system

In NPD and CANDU and other reactors being planned, the high-pressure coolant is contained within Zircaloy pressure tubes surrounding the fuel. Failure

of a pressure tube is less serious than failure of a reactor pressure vessel would be, provided that failure of one tube does not cause the failure of others. Two series of tests have been conducted to investigate the proviso.

One series [13] studied the failure of a pressure tube at normal operating conditions using a full-scale mock-up of a section of the NPD and CANDU cores. These tests showed that the adjacent tubes will not be damaged if a pressure tube fails at normal operating conditions.

The other series of tests is investigating the failure of a pressure tube that might result from a nuclear power excursion. The coolant is approximately at critical temperature and pressure (370 °C, 225 kgf/cm²) and the failures are caused by a thermite charge used to simulate a melting fuel bundle. In the tests performed, the adjacent pressure tubes have not failed although some were bent by the shock wave.

A safety factor of three between the design stress and the ultimate tensile strength of the Zircaloy pressure tubes has been accepted because they are of simple shape which permits accurate stress analysis, close control is exercised over their manufacture and installation, and there is considerable experience with Zircaloy [14]. Although the ultimate strength increases with exposure to radiation, the design is based on the strength of the unirradiated material. One of the pressure tubes in NPD was examined with special equipment seventeen months after start of operation. No undesirable effects were found [15].

Fuel

The fuel for the power reactors is natural UO_2 clad in Zircaloy. Extensive research and development has led to the present design of 19-element bundles 49.5 cm long [16–18].

The fuel rating is governed by the maximum acceptable operating temperature of the UO_2. This is limited by thermal effects such as expansion, cracking and melting, and by fission-gas release which increases rapidly at temperatures exceeding 1 700 °C. The rating of the power reactor fuel has been limited to $\int \lambda d\theta = 40$ W/cm which gives a maximum UO_2 temperature of 1 900 °C. Experimental irradiations have shown that the fuel can be operated at this rating with a high degree of confidence.

The acceptable fuel rating is limited also by the cladding temperature that would result from a sudden loss of cooling. If cooling is lost, the temperature of the cladding will rapidly approach the mean fuel temperature and then will rise at a rate determined by the continuing heat generation. Assuming a simultaneous loss of cooling and a reactor shut-down, the cladding of the above fuel would reach 1 100 °C in 5 seconds and 1 820 °C (the melting point of Zircaloy) in 2½ minutes. In CANDU, it is extremely improbable that cooling would be removed from the fuel in less than 2 seconds even in the worst loss of coolant. In that event, the cladding temperature would not reach 1 100 °C until 20 seconds after the accident and 1 820 °C until 3½ minutes. For most assumed failures of the primary cooling system, some cooling of the fuel will continue much longer than 2 seconds and the emergency cooling system can be fully operative and effective before any fuel failure occurs. CANDU-type fuel has been tested at 1 100 °C by pressurizing internally with a quantity of gas five times that calculated to be released from the UO_2 without significant failure.

Corrosion and allowable stress for the pressure tubes were dominant factors in choosing the coolant flow and temperature conditions. The resulting minimum burn-out heat flux for the CANDU fuel is approximately 2.5 times the normal heat flux.

PROTECTIVE DEVICES

Emergency cooling system

After a loss of cooling the temperature of the fuel sheathing would rise. Cooling should be re-applied before the temperature of the cladding reaches the failure point.

The emergency cooling system is designed to prevent significant damage to the fuel in the event of any single rupture in the primary coolant system outside of the reactor. If a pressure tube fails, the emergency cooling system will prevent failure of fuel elements in other pressure tubes. The system is designed so that important parts such as valves can be tested periodically. Flywheels are added to the pumps and the system is designed to permit thermal convection to avoid loss of effective cooling resulting from loss of power to the coolant system pumps.

The protective shut-down system

The automatic protective shut-down system in Canadian reactors is designed to prevent overheating of the fuel cladding in the event of any single conceivable failure in the process equipment. The worst conceivable malfunction in the regulatory system is an increase in reactivity at the most rapid rate that is mechanically possible for the regulating system. The worst conceivable failure in the primary cooling system is a complete severance of the largest pipe in that system while the reactor is operating at full power.

After the 1952 accident, NRX was re-designed to permit dumping of the heavy-water moderator as a back-up to the shut-off rods. In NPD and CANDU, rapid dumping of the moderator is the only protective action.

The NRX accident also gave rise to the concept of a shut-down reserve or "safety bank" [2]. In the power reactors the practice of returning part of the moderator to the reactor after dumping ensures that shut-down capacity is always available.

Redundancy is used to increase reliability and permit frequent testing of shut-down devices. The protective systems have three independent channels operating in a two-out-of-three coincidence. Each channel can be tested from its sensing devices through to the moderator dump valves, at any time during operation without shutting down the reactor, as often as necessary to maintain the desired degree of reliability.

There is redundancy also in the number of conditions that will cause an automatic shut-down. For every conceivable process-system failure, changes in at least two physical variables would produce signals that would stop reactor operation.

An unreliability of less than 0.003 is easily achieved with daily testing if only independent faults occur. One cross-linked fault has been reported in the NPD system. This occurred when the sensing devices of one type in each of the three channels were rendered inoperative at the same time by error during maintenance while the reactor was shut down. It revealed itself during the attempt to bring the reactor back into operation because the fault was fail-safe. However, since one cross-linked fault may overshadow independent faults in an analysis of unreliability, the incident is a reminder that great effort must be made to reduce cross-linked faults.

CONTAINMENT

Containment provisions have been considered necessary for the power reactors. They are designed to give protection of the public for the worst conceivable single rupture of the piping of the reactor cooling system.

In NPD, pressure relief and pressure suppression by dousing are provided to obviate the need for a large containment building. The reactor and primary system are enclosed in heavily shielded vaults below ground with a large duct opening to the atmosphere. The duct, which is normally closed by a light diaphragm, would release the intitial discharge of steam from any failure in the primary system. Then it would be closed by a door before fuel failure occurred.

The size of duct needed to prevent damage to the vaults by the pressure transient caused by possible failure of the primary system was determined by two methods of shock-wave analysis [19, 20] and model studies on 1/20 and 1/4 linear scales.

Larger-scale tests [21] simulated the entire arrangement including coolant system, containment vault and duct. Guided by these tests, the duct was made 2.7 by 3.6 m in cross section and 15 m long. It leads from a vault of approximately 2 160 m^3 in which there is a coolant system containing 10 tonnes of water at approximately 260 °C.

As part of the containment provisions of NPD and Douglas Point, a large water dousing spray is provided to condense the steam formed by the coolant which would be discharged at high enthalpy. A full-scale section of the Douglas Point dousing region was built to test the concept.

A difficult problem in designing containment provisions is to determine the rate of loss of coolant through any assumed break in the cooling system. Analysing the discharge for a pressure-tube system is difficult because it is transient since there is no large reservoir of coolant. Experiments were conducted, including some in which a thermal gradient was provided along the pipe to reproduce operating conditions [22–24]. With the temperature gradient existing in Douglas Point (249 °C inlet, 293 °C outlet) the discharge rate from a ruptured pressure tube could exceed 6.3 kg/cm^2s.

The containment of the Douglas Point plant employs a cylindrical concrete building designed to withstand 0.4 kgf/cm^2, with a dousing arrangement to prevent the pressure from reaching this level. The specified leakage rate is 0.1 % hour at 0.4 kgf/cm^2. Since the pressure can be reduced within an hour using the ventilation-system coolers, the ineffectiveness of the containment is taken at 0.001, i.e., the total leakage for the containment after an accident is taken as 0.001 of the contained activity.

SITING

General

No rules have been established in Canada for the siting of nuclear reactors. While a very safe 1 MW pool reactor has been approved for a university campus in a large city, the sites approved for power reactors have been in regions of low population.

It is possible to design and construct a nuclear plant so that it can be operated in any location without exceeding the recommendations of the ICRP [25] regarding permissible exposure of the public. Therefore, sites have been judged primarily by considering possible accidents.

It is difficult to limit the inhalation of fission products by the public after their escape. This risk to the publie depends on the density of population and its distance from the reactor. The inhalation of radioactive iodine is the most important factor to be considered in the selection of a site.

Exclusion area

The public is excluded from the land close to nuclear reactors. The exclusion area is intended to protect the public in any accident where process equipment and protective devices fail but containment provisions do not.

The authors feel that the exclusion area should be so large that in such an occurrence no-one outside of it would receive more than 25 rem to the whole body and 250 rads to the thyroid except under extremely infrequent weather conditions. For example, with a release of 1% of the radioiodine from the fuel and a containment ineffectiveness of 0.001, at most 10^{-5} of the radioiodine would escape. The shortest distance from the Douglas Point reactor to the boundary of the exclusion area is 900 m, and from the NPD reactor it is 1 100 m. Under very unfavourable weather conditions (a dilution rate of 10^{-4} m^3/s [26]), the iodine exposure at the exclusion boundary of the Douglas Point reactor with this release would be of the order of 0.1 cs/m^3 which would give about 250 rads to the thyroid of an infant.

Records have been kept of atmospheric contamination at a distance by the normal release of argon-41 from the Chalk River stacks. They provide useful information for predicting the dispersal and dilution of accidental releases [26, 27].

Population density outside the exclusion area

If both protective devices and containment provisions were defective when a dangerous fault occurred, the quantity of radioactive material escaping to the atmosphere might be many times that given above and serious over-exposures could result. With good maintenance, supervision and discipline the unreliability of the containment provisions should not exceed 0.003. Consequently, the probability of this accident should be two and a half orders of magnitude less than that of the accident in which containment does not fail.

The consequences of such disasters depend on the density of population, its distance from the reactor, and meteorological conditions. In such a disaster the probability of being exposed to a dangerous concentration decreases rapidly with an increase in r, the distance from the reactor in km. To give greater importance to population close to the reactor when comparing different sites, the population density, p, has been multiplied by a weighting factor, $W = r^{-3}$. A population index, J, has been defined by

$$J = \int Wp\mathrm{d}A$$

integrated over the land area, A, outside the exclusion area [28]. For most of the older nuclear power stations, J is between 100 and 1 000. For CANDU and NPD, it is less than 100. In a densely populated city, it would be over 100 000.

TESTING AND INSPECTION

Testing

Consideration of safety of a reactor does not end with analysis of the design [29]. Equipment must be operated as intended. Frequent tests and obervations must be made to verify that the equipment and systems perform as assumed and to ensure that their effectiveness and reliability are maintained.

During the commissioning of the plant, tests adequate to prove the effectiveness of the protective devices and the containment provisions and the general safety of the nuclear plant must be prepared and performed thoroughly and competently, and the results must be properly recorded.

The operators are expected to keep adequate records of maintenance and of all faults. These call attention to the possible causes of accidents, indicate desirable changes or improvements and provide a continuing record of the unreliability of the equipment. The fault records also provide a basis for estimating the reliability of similar equipment proposed for future plants.

The fault record of the protective systems will determine the required frequency of testing. Since the independent triplicated protective system developed and required on Canadian power reactors can be tested while the reactor is operating, the frequency of testing can be governed by the observed fault rate.

Inspection

Besides the normal inspection required by the owner of a reactor to control manufacture and construction there is independent inspection done by the staff of the Atomic Energy Control Board to verify that regulations, approved design features and operating procedures are followed. The inspection is valuable in encouraging the development of safe operating techniques and the improvement of regulating and protective equipment, and in promoting interest in reactor safety.

Reactor inspections begin during construction, increase in frequency to the extent that a nuclear-safety engineer is temporarily attached to the commissioning staff, and are less frequent after full-power operation has been authorized.

CONCLUSIONS

It is now realised that the risk of disastrous reactor accidents is not nearly as great as it was imagined to be a few years ago. Experience of accidents and experimental investigation have shown that more than 10^5 curies of fission products is not easily released from the fuel of a large reactor. In more than 1 000 reactor-years of operating experience throughout the world, there have been less than ten occasions that the air in a reactor building was dangerously contaminated by the accidental release of fission products [30]; i.e., their frequency has been less than 10^{-2} per reactor-year. The unreliability of containment provisions can be reduced below 10^{-3}, thereby reducing the expected frequency of dangerous contamination of air outside the building. The probability that this contamination

will have dangerous concentration when it has drifted beyond the exclusion area is small. Therefore, the probability of an accident in a reactor, during its useful life, that causes dangerous over-exposure of persons outside the exclusion area is extremely small.

As the reliability and effectiveness of safety measures increase and the good safety record of reactors is maintained or improved, we may expect them to be located in more populated regions than they are at present.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/28 Canada

Pratique et expérience touchant la sûreté des réacteurs au Canada

par G. C. Laurence *et al.*

Au Canada, la réglementation dans le domaine de l'énergie atomique concernant l'intérêt public est établie par la Commission de contrôle de l'énergie atomique. Pour ce qui a trait à la sûreté des réacteurs, la Commission s'acquitte de sa responsabilité avec le concours et les conseils des Ministères fédéraux et provinciaux de la santé et du travail et d'un Comité consultatif pour la sûreté des réacteurs. L'opinion et la pratique à ce sujet s'inspirent fortement de l'expérience acquise et des recherches effectuées dans les laboratoires d'Atomic Energy of Canada Limited.

L'expérience et les essais ont abouti à des principes tels que: « la réserve de sûreté » ou réserve d'arrêt, l'emploi d'une surabondance de systèmes de contrôle pour permettre de dépister et de corriger rapidement les défectuosités, l'emploi du contrôle par niveau du modérateur avec réservoir de vidange pour les arrêts d'urgence, l'indépendance des chaînes de réglage et de protection, la nécessité de deux signaux d'arrêt indépendants pour le cas des défectuosités éventuelles dans les systèmes de contrôle, la prévision d'une réserve

adéquate de l'agent réfrigérant pour le refroidissement d'urgence du combustible, et la condition que les dispositifs d'urgence ou d'appoint ne restent pas longtemps inactifs sans être soumis à une épreuve. L'application de certains de ces principes depuis la deuxième Conférence de Genève est exposée dans le mémoire.

Au stade de l'étude des risques associés aux réacteurs, on consacre une étude plus poussée aux appareils dont les risques de pannes éventuelles pourraient donner lieu à des conséquences graves, plutôt que de calculer le nombre d'accidentés ou autres conséquences pouvant résulter du « pire accident croyable »; cette façon de procéder dirige l'attention sur les causes d'accidents et, par conséquent, favorise l'adoption de meilleures mesures préventives. Un intérêt croissant est apporté à la détermination du degré d'incertitude de fonctionnement de l'équipement des réacteurs par des observations sur les appareils en service.

Il est tenu pour particulièrement important que les systèmes de fonctionnement (y compris le réglage de la réactivité), les dispositifs automatiques de protection, et les mesures de rétention, soient indépendants les uns des autres afin que des pannes simultanées dans deux secteurs quelconques attribuables à la même cause soient peu probables. Les dispositifs automatiques de protection doivent être tels qu'une panne dans le système de réglage ne puisse faire des victimes à l'intérieur ou à l'extérieur de l'usine. Les mesures de rétention doivent être conçues de façon à protéger les personnes à l'extérieur de l'usine, si le système de réglage ou un des systèmes de fonctionnement et les dispositifs automatiques de protection tombent en panne simultanément. Toutes les installations doivent être conçues de sorte que les défauts dans les parties ou les pièces constituantes puissent être détectés peu après leur apparition.

On a exécuté beaucoup de travaux d'expérimentation et de mise au point dans le domaine des mesures de rétention tels que la suppression des vapeurs par giclement d'eau, et la fermeture automatique de l'enceinte de rétention à la suite de tout accident qui entraîne une perte d'agent réfrigérant avant que le combustible ne soit surchauffé par l'activité des produits de fission. Parmi les autres problèmes de sûreté qui ont fait l'objet d'expériences, citons les conséquences d'une rupture accidentelle d'un tube de force à l'intérieur du réacteur, le surchauffage des cartouches de combustible, le comportement des produits de fission gazeux dans l'oxyde surchauffé, et la dispersion des produits de fission lorsqu'ils sont libérés par accident dans l'atmosphère, dans le sol, dans les fleuves et les lacs.

Les progrès en ce qui concerne la sûreté des réacteurs et l'expérience continue d'un fonctionnement sûr des installations militent en faveur de l'aménagement de réacteurs dans des régions un peu plus peuplées qu'auparavant.

A/28 Канада

Опыт эксплуатации по безопасности реакторов в Канаде

Г. К. Лоренс *et al.*

В Канаде Управление по контролю над атомной энергией является директивным органом, постановления которого в области атомной энергии обязательны. Свои обязанности по обеспечению безопасности реакторов это Управление выполняет при помощи и консультировании со стороны общегосударственного и провинциальных управлений здравоохранения и труда, а также Консультативного комитета по безопасности реакторов. В Канаде на решения проблем безопасности реакторов в сильной степени влияют опыт и научно-исследовательские работы в лабораториях компании «Атомик энерджи оф Канада».

Опыт и эксперименты привели к формированию таких принципов, как «блок безопасности» или резерв на выключение; использование статических неопределенностей в системах регулирования, позволяющее быстро найти и исправить ошибки; использование регулирования уровня замедлителя, включая главное устройство для аварийного выключения реактора; независимость регулирующей и защитной систем; необходимость в двух независимых улавливателях во избежание любой ошибки в регулировании; обеспечение соответствующего количества аварийного теплоносителя для топлива; требование, чтобы аварийные или резервные устройства не оставались длительное время в бездействии или непроверенными. Описано развитие некоторых из этих принципов со времени Второй Женевской конференции.

При оценке опасностей, связанных с реакторами, вероятности повреждений узлов оборудования, которые могут привести к тяжелым последствиям, уделяется больше внимания, чем рассмотрению числа несчастных случаев и других последствий «самой серьезной возможной аварии», так как это привлекает внимание к причинам аварий и тем самым способствует увеличению безопасности. Усилено внимание к определению ненадежности реакторного оборудования путем наблюдений за эксплуатацией.

Особо важным считается то, что технологические системы (включая регулирование реактивности), автоматические защитные устройства и системы удержания должны быть независимы друг от друга, так что одновременное повреждение любых двух из них по одной и той же причине очень маловероятно. Автоматическое защитное оборудование должно быть таким, чтобы любое повреждение в системе регулирования не могло вызвать поражения персонала, работающего на установке, и окру-

жающего населения. Безопасные условия удержания должны быть разработаны таким образом, чтобы предотвратить поражение лиц, находящихся за пределами предприятия, даже в случае одновременного выхода из строя системы регулирования или другого технологического оборудования и автоматических защитных устройств. Все оборудование должно быть сконструировано таким образом, чтобы повреждения в деталях или узлах могли быть обнаружены сразу и немедленно устранены.

Серьезные экспериментальные и опытно-конструкторские работы проводятся по созданию безопасных условий удержания, например подавление пара распылением воды и автоматическое закрытие защитной оболочки в случае прорыва контура теплоносителя, прежде чем топливо перегреется вследствие распада активности продуктов деления. Среди других исследованных проблем безопасности следует назвать: повреждение вследствие аварийного разрыва трубы давления с теплоносителем в реакторе, перегрев тепловыделяющих элементов, поведение газообразных продуктов деления в перегретом топливе в виде окиси, распыление продуктов деления, выделившихся в случае аварии, в атмосфере, на почве, в воде.

Успехи, достигнутые в области безопасности реакторов, и накопленный опыт безопасной эксплуатации вселяют надежду, что реакторы можно будет строить в несколько более населенных районах, чем это делалось ранее.

A/28 Canadá

La experiencia y los métodos de seguridad de los reactores en el Canadá

por G. C. Laurence *et al.*

En el Canadá, la Atomic Energy Control Board es el organismo ordenador de las actividades nucleares. En cuanto a la seguridad de reactores, dicho organismo se libera de sus responsabilidades con la ayuda y el asesoramiento de los departamentos nacionales y provinciales de trabajo y de sanidad, y de la Comisión Asesora de Seguridad de Reactores. Las ideas y los métodos canadienses con respecto a la seguridad de los reactores han sido muy influídos por la experiencia y las investigaciones llevadas a cabo en los laboratorios de la compañía Atomic Energy of Canadá, Ltd.

La práctica y la experiencia han hecho formular principios como los siguientes: los « venenos de seguridad » o reserva de apagado, el uso de circuitos múltiples en los sistemas de control para localizar y corregir con rapidez las averías, el control por nivel del moderador con vaciado rápido combinado para el apagado de emergencia, la independencia de los sistemas de control y de protección, la necesidad de dos señales de parada independientes para el caso de una posible avería en el control, provisión de una adecuada refrigeración de emergencia del combustible y el requisito de que los dispositivos de emergencia o de reserva no permanezcan parados o sin comprobar durante mucho tiempo. Se describe la evolución positiva que han sufrido algunos de estos principios desde la Segunda Conferencia de Ginebra.

Al evaluar los riesgos de los reactores, se presta mayor consideración a la probabilidad de fallos del equipo que puedan desencadenar consecuencias graves, que al número de víctimas y a otras consecuencias del « accidente máximo previsible », porque así la atención se fija en las causas de los accidentes, lo que estimula el progreso de la seguridad. Cada día se presta más atención a la estimación del grado de confianza que merece el equipo del reactor mediante la observación de su funcionamiento real.

Se considera de importancia especial el que los sistemas de proceso (incluido el control de la reactividad), los dispositivos automáticos de protección y las barreras de contención sean independientes entre sí, de modo que sea muy pequeña la probabilidad de que fallen por la misma causa dos cualesquiera de dichos sistemas o dispositivos. El equipo automático de protección debe ser tal que no se espere que ninguna persona situada dentro o fuera de la instalación sufra daño como resultado de un fallo del sistema de regulación. Las barreras de contención deben proyectarse para que eviten los daños a la población ajena a la instalación en el caso de que fallen simultáneamente el sistema de control u otro equipo de proceso y los sistemas automáticos de protección. Todo el equipo debe proyectarse para que las averías en las piezas o componentes puedan ser descubiertas poco después de que tengan lugar y ser reparadas sin demora.

Sobre las barreras de contención se han llevado a cabo muchos programas experimentales y de desarrollo, tales como la relajación de la presión por aspersión de agua y el cierre automático de la vasija de contención antes de que la desintegración radiactiva de los productos de fisión pueda sobrecalentar el combustible, en el caso de accidente por pérdida del refrigerante. Entre otros, se han investigado además los siguientes problemas de seguridad: daños que resultan de la ruptura accidental de un tubo de presión del reactor, el sobrecalentamiento de los elementos combustibles, el comportamiento de los productos de fisión gaseosos en óxido de uranio sobrecalentado, y la dispersión de los productos de fisión, liberados accidentalmente en la atmósfera, en el terreno y en los ríos y lagos.

Los progresos en la seguridad de los reactores y la experiencia acumulada en la explotación segura de los mismos animan ya a localizar los reactores en regiones algo más densamente pobladas que al principio.

Conception de la sûreté en France et influence des impératifs de sûreté sur la conception des réacteurs

par F. de Vathaire, P. Vernier et A. Pascouet *

La sûreté des réacteurs est l'un des secteurs de l'énergie atomique pour lesquels la collaboration internationale peut se développer d'une manière particulièrement favorable. C'est en effet l'intérêt général de tous les pays de mettre en commun les résultats acquis dans ce domaine (études de sûreté, expérience d'exploitation, comptes rendus d'incidents), afin de réduire les risques d'accidents qui pourraient avoir des conséquences psychologiques défavorables au développement de l'énergie atomique.

Beaucoup a déjà été réalisé dans ce domaine: colloque de l'AIEA de mai 1962, revue *Nuclear Safety*, collaboration bilatérale entre divers pays, etc. Pour sa part, la France a présenté en 1962 ses conceptions en matière de sûreté [1–7] et elle échange régulièrement des informations sur les études de sûreté des réacteurs avec la Grande-Bretagne (notamment sur les réacteurs graphite-gaz et les réacteurs à neutrons rapides) et avec les Etats-Unis (plus particulièrement sur les réacteurs à eau et à neutrons rapides).

Les réacteurs de la filière graphite-gaz constituant l'élément essentiel du programme français de réacteurs de puissance, un mémoire spécial ** leur est consacré. Un autre mémoire *** présente le réacteur CABRI, dont la récente entrée en service permettra de développer en Europe les études de sûreté des réacteurs à eau. Le présent mémoire se propose d'exposer les études concernant les divers types de réacteurs (autres que les réacteurs graphite-gaz) qui sont actuellement développés en France: réacteurs eau lourde-gaz (type EL4), réacteurs à neutrons rapides (type RAPSODIE), réacteurs de recherche à eau.

Les centrales à eau sous pression ou à eau bouillante ne font pas actuellement l'objet d'études de sûreté, mais la France, qui participe à la centrale de Chooz, du type eau sous pression, suit avec intérêt l'important programme d'études de sûreté que l'AEC développe dans ce domaine.

CONCEPTION GÉNÉRALE DE LA SÛRETÉ EN FRANCE

L'objectif est de réaliser des réacteurs présentant un très haut degré de sûreté: la probabilité d'accidents graves doit être réduite à un niveau suffisamment faible pour être acceptable, et les conséquences de tels accidents du point de vue de la contamination radioactive doivent être suffisamment limitées pour ne pas mettre en jeu la sécurité du personnel et des populations avoisinantes. Différents stades peuvent être distingués dans l'étude de la sûreté d'un réacteur. Au stade de l'avant-projet, la validité des principes (sens circulation, densités de puissance, etc.) est vérifiée, l'étude des transitoires accidentels est effectuée, et les essais qui peuvent apparaître nécessaires pour préciser certains points sont réalisés; on vérifie que les dispositifs d'arrêt et de confinement proposés limitent efficacement les conséquences de ces transitoires. Au stade de la construction et des essais, la sécurité propre des circuits et des appareils, la mise en œuvre des matériaux, les performances réelles font l'objet d'études et de contrôles minutieux. Au stade de l'exploitation, la qualification du personnel, l'exactitude du mode d'emploi, la prévention des incidents et leurs remèdes sont soigneusement examinés. A tous les stades, l'esprit « sûreté » est développé au moyen de contacts fréquents entre ingénieurs chargés du réacteur et ingénieurs chargés du contrôle de la sûreté.

Sur le plan technique, le confinement des produits de fission est assuré au moyen des trois barrières classiques: la gaine, le circuit de refroidissement primaire et, si elle existe, l'enceinte munie de ses dispositifs de filtration. Les accidents possibles sont d'une part les accidents de refroidissement, d'autre part les accidents de réactivité. Dans la plupart des types de réacteurs, les problèmes de sécurité les plus importants sont liés aux accidents de refroisissement. Le plus souvent, des précautions suffisantes peuvent être prises contre les accidents provenant des variations de vitesse du réfrigérant, bien que les solutions adoptées pour la disposition des boucles de refroidissement et pour les moyens de soufflage puissent être variées. Par contre, la perte de réfrigérant par dépressurisation, surtout lorsque celle-

* Commissariat à l'énergie atomique, avec la collaboration de P. Candes, G. Drevon et A. Meunier.

** Voir, dans le présent volume, P/81.

*** Voir, dans le présent volume, P/103.

ci est brutale, pose des problèmes qui nécessitent souvent des calculs et des essais permettant d'estimer les dégâts des structures, l'évolution de température du combustible et la libération des produits de fission. Dans presque tous les cas, les conséquences de tels accidents sont essentiellement d'ordre économique (endommagement d'une charge de combustible, contamination de certains circuits ou locaux, etc.), tandis que la santé des populations n'est pas mise en jeu. Les accidents de réactivité n'entraînent pas de dangers sensibles dans le cas des réacteurs à uranium naturel (modérateur graphite ou eau lourde); pour les réacteurs à combustible très enrichi (réacteurs à eau ou à neutrons rapides), ces accidents doivent être pris en compte, car, bien que leur probabilité puisse être le plus souvent rendue très faible, des erreurs humaines sont toujours possibles. L'étude des accidents de réactivité peut nécessiter des réacteurs spéciaux (réacteurs SPERT, CABRI, etc.). La détermination de leurs conséquences mécaniques peut être effectuée au moyen de maquettes, où l'excursion de puissance est simulée au moyen d'explosifs chimiques: deux exemples de telles simulations sont indiqués ci-après (réacteurs à neutrons rapides et réacteurs à eau).

ÉTUDES DE SÛRETÉ RELATIVES AUX RÉACTEURS EAU LOURDE-GAZ

Les caractéristiques adoptées pour les réacteurs modérés à l'eau lourde et refroidis au gaz développés en France sont exposées dans d'autres communications. La philosophie de la sûreté de ce type de réacteur est conforme à ce qui a été indiqué au paragraphe précédent. Ce sont surtout les accidents de dépressurisation qui ont nécessité l'exécution d'essais destinés à évaluer les dégâts mécaniques et le comportement du combustible; le programme de ces essais et quelques résultats d'essais réalisés sont indiqués ci-après [6, 8, 9].

Rupture d'un tube de force

a) *Conséquences d'une rupture brutale sur les structures de la cuve.* – Les résultats des essais d'explosion de tubes de force faits sur modèles au 1/10 sont exposés en référence [8]; celle-ci indique de façon détaillée l'influence de divers paramètres (masse de gaz, section de rupture, niveau de l'eau, élasticité de la cuve, etc.). Les conséquences pratiques sur la construction de la cuve sont importantes: citons entre autres la pression de calcul de la cuve et le système d'évacuation et de récupération de l'eau lourde expulsée par l'explosion.

b) *Conséquences de la rupture sur le combustible.* – Les effets mécaniques de la rupture sur l'intégrité du combustible dans le canal qui explose feront l'objet d'essais sur le modèle d'un canal hors pile à l'échelle 1 vers la fin 1964.

c) *Risque de dénoyage.* – Une baisse accidentelle du niveau d'eau lourde dans la cuve peut découvrir un tube de force; celui-ci est alors soumis à des gradients de température qui peuvent le déformer et éventuellement entraîner son flambage. Des essais sur canal réel dans la géométrie du réacteur EL4 ont montré que le flambage n'était pas possible dans des conditions de charge mécanique du canal jugées pourtant pessimistes, et qu'il n'y avait aucun risque de rupture.

Rupture d'un petit collecteur alimentant un tube de force

Les effets d'un transitoire thermique sur le combustible du canal (ou sur l'ensemble du combustible) du réacteur sont peu importants; par contre, les efforts aérodynamiques, particulièrement sur la dernière grappe du canal, sont importants, mais n'entraînent probablement pas la destruction du combustible; les calculs étant insuffisants pour apprécier la tenue du combustible à ces efforts, des essais sur maquettes, analogues aux essais du paragraphe b) ci-dessous, seront faits en 1964.

Rupture d'un gros collecteur

a) *Effets thermiques sur le combustible.* – Des programmes de calcul permettant de calculer le débit de gaz en régime transitoire et l'évolution correspondante des températures ont été mis au point et vérifiés expérimentalement. Le choix des hypothèses faites sur l'accident (emplacement et section de la rupture donnant un débit nul, ou au contraire rupture de dimension maximale située à un endroit du circuit jugé critique) et les valeurs numériques choisies pour les calculs peuvent influer notablement sur les résultats. Certaines valeurs paramétriques mal connues font l'objet de déterminations expérimentales: c'est le cas par exemple des pertes thermiques par rayonnement dans une grappe aux températures élevées et surtout de l'évaluation de la pression dans les gaines due aux gaz de fission.

La température de fusion des gaines n'étant jamais atteinte dans ce type de transitoire, même le plus pessimiste, avec l'isolement thermique choisi, un critère de rupture de gaine doit être recherché; le fluage dû à la pression intérieure aux températures élevées (900 à 1 300 °C environ) peut être la cause de ces ruptures; des essais ont été effectués et les résultats préliminaires concernant ce point sont exposés en référence [9]. Un four spécial est en construction pour effectuer des mesures plus systématiques sur différents types de gaines (aciers, zirconium allié ou béryllium); il est particulièrement intéressant de connaître l'augmentation du volume libre d'un crayon avant rupture.

Enfin, une boucle spéciale permettant des dépressurisations brutales sera utilisée dans le réacteur PÉGASE au début 1965; elle permettra d'effectuer des vérifications globales du comportement d'une grappe complète de combustible aux différents transitoires envisagés.

b) *Efforts aérodynamiques.* – Ils peuvent être approximativement calculés (compression sur la chemise de graphite, traction sur les grilles des grappes) soit durant la remontée des ondes de détente, soit durant le régime semi-permanent de dégonflage. Cependant une vérification expérimentale apparaît nécessaire, car il est essentiel de s'assurer que le combustible n'est pas détruit pendant les quelques secondes de dégonflage et que la libération des produits de fission se produit, nettement après la fin du dégonflage, par suite des effets thermiques. Des essais sur maquettes auront lieu au milieu de 1964.

Enceinte de sécurité

Les hypothèses faites sur l'accident de perte de réfrigérant primaire déterminent le type d'enceinte et peuvent justifier ou non l'existence même de l'enceinte. Dans le cas de petites ruptures (tubes de force, collecteurs indivuduels), les fuites de gaz doivent pouvoir être canalisées par les enceintes existantes (cuve, protection biologique) vers les filtres, au besoin en passant par une chambre de refroidissement de façon à conserver l'efficacité des filtres. Dans le cas de la rupture d'un gros collecteur, la température du gaz et les efforts aérodynamiques ne doivent pas endommager les filtres de l'enceinte; si l'on ferme l'enceinte au moment de l'accident par des manœuvres de vannes, l'enceinte doit pouvoir tenir à une certaine pression et à une certaine température, qui peuvent être abaissées par un système de douche avant l'évacuation ultérieure des gaz à travers les filtres. Si l'enceinte reste «ouverte», il nous semble possible de protéger le système de filtration sans effectuer des manœuvres immédiates de vannes et de laisser la ventilation initiale en fonction durant le dégonflage du circuit primaire. Enfin, suivant les dimensions ou la disposition du réacteur, un système de double enceinte, dont l'une tient la pression et l'autre récupère les fuites, est également envisageable.

Les essais à effectuer concernent essentiellement la tenue des filtres aux conditions accidentelles [tenue mécanique et tenue au milieu (température, humidité, poussières, vapeurs)]. Par ailleurs, des essais d'efficacité de la douche de l'enceinte d'EL4 sont en cours.

ÉTUDES DE SÛRETÉ RELATIVES AUX RÉACTEURS RAPIDES REFROIDIS AU SODIUM

Les préoccupations qui sont à la base des études de sûreté de la filière des réacteurs rapides refroidis au sodium ne sont pas différentes de celles qui s'appliquent aux réacteurs des autres filières. Deux sortes d'accidents doivent être évités, l'accident de réactivité et l'accident de refroidissement, et, dans la mesure où l'on ne peut être certain de rendre nulle leur probabilité, on cherche à réduire au minimum les dommages qu'ils pourraient provoquer.

Accidents de réactivité

A priori, le court temps de vie des neutrons dans les réacteurs rapides les défavorise puisque, pour un excès de réactivité effectif un peu supérieur au dollar, leur période est nettement plus courte que celle des réacteurs thermiques. Aussi est-ce un domaine dans lequel on doit s'interdire de pénétrer dans tous les cas. D'où l'intérêt d'une forte contre-réaction négative pour ces réacteurs.

Lorsqu'il s'agit de réacteurs rapides de grande taille (cœur de l'ordre du millier de litres et plus) refroidis au sodium, dont la composition présente un intérêt économique, la réalisation de cet objectif se heurte à quelques difficultés; le coefficient de température des combustibles économiquement satisfaisants (métal à gaine épaisse et céramique) est mal connu. Par ailleurs, le coefficient lié à la densité des noyaux de sodium tend à être positif au centre des cœurs qui ont des fuites faibles. Sa valeur est en outre mal précisée. Par contre, dans ces réacteurs à spectre mou qui contiennent beaucoup de matériau fertile, on peut compter sur un effet Doppler négatif et grand. Bien que la valeur de ce coefficient soit également mal connue, il semble que, dans les cas d'incidents réalistes, l'effet d'un coefficient sodium positif puisse être compensé et au-delà par celui d'un coefficient Doppler négatif.

Dans l'état actuel des connaissances, il y a tout lieu de penser qu'il sera techniquement possible d'assurer la protection des réacteurs rapides contre l'instabilité et les accidents de réactivité, mais que les incertitudes qui subsistent encore sur les données de base amèneraient actuellement à prendre des marges de sécurité économiquement coûteuses (par exemple sacrifice sur le taux de surgénération pour avoir un coefficient sodium satisfaisant). Toutefois, on peut penser que, d'ici la fin de la prochaine décennie (moment où les centrales nucléaires rapides de taille industrielle devront entrer en service), ces incertitudes seront levées.

Accidents de refroidissement

L'insuffisance du refroidissement peut entraîner des dommages pour le combustible ou les structures du réacteur et éventuellement provoquer un accident nucléaire par déformation du cœur. Il faut noter que les fortes puissances spécifiques des réacteurs rapides accélèrent les évolutions de température qui accompagnent un déséquilibre entre puissance et refroidissement. Les accidents dus à un arrêt général du refroidis-

sement (arrêt des pompes, perte de fluide refroidisseur) peuvent être éliminés par des dispositions du projet (nombre et inertie des pompes, double enveloppe des circuits, réserve de fluide). La prévention d'accidents de refroidissement locaux (dus à des perturbations au niveau des aiguilles de combustible) pose un problème plus difficile. Un effort doit être fait sur la détection rapide de ces incidents dont le développement clandestin est particulièrement dangereux. La possibilité d'une réaction sodium-eau dans les générateurs de vapeur doit également être examinée attentivement.

Etudes de sûreté effectuées en France

Les études de sûreté effectuées par le CEA dans le domaine des réacteurs rapides ont été orientées jusqu'ici principalement vers la solution des problèmes qui se posaient pour la mise au point du projet RAPSODIE. Parmi ceux-ci, le plus important et le premier à résoudre concernait la définition des enceintes de sûreté de la pile. L'étude des enceintes de sûreté a conduit immédiatement à considérer que les conditions les plus graves dans lesquelles ces enceintes devraient jouer leur rôle étaient une libération brutale d'énergie dans la pile et un feu de sodium. Comme il est apparu également que les feux de sodium de beaucoup les plus dangereux étaient ceux qui étaient susceptibles de se produire à la suite de libérations explosives d'énergie, l'étude des enceintes de sûreté de RAPSODIE, menée dans le cadre de l'Association EURATOM-CEA « neutrons rapides », a été axée sur l'étude des phénomènes mécaniques accompagnant une excursion de puissance éventuelle dans la pile.

Cette étude [2, 11, 12], réalisée avec la collaboration du Service des poudres, simule au moyen d'explosifs chimiques (hexogène CH notamment), des dégagements d'énergie allant jusqu'à 1, 15×10^8 cals ($4,81 \times 10^8$ Js) libérées en 80 μs à l'échelle 1. Une série de 11 explosions a été réalisée sur une maquette à l'échelle 3/10 représentant de façon assez fidèle la pile, le sodium étant toutefois remplacé par de l'eau. Une trentaine d'essais avec du sodium sur une maquette à l'échelle 1/10 sont envisagés pour étudier la validité de la simulation par l'eau *. Une équipe de la UKAEA a apporté son concours à ces expériences en effectuant des mesures des pressions dynamiques. Les résultats feront l'objet d'un rapport en préparation. L'étude de la validité de la simulation se poursuit.

Ces études expérimentales ont contribué à définir les caractéristiques du bloc pile de RAPSODIE et notamment à concevoir un dispositif limitant le soulèvement des bouchons en cas d'accident et éliminant toute projection de matériaux ou de mécanismes. Elles permettent en définitive de dire que, dans les conditions très pessimistes de l'excursion nucléaire maximale calculée, l'enceinte formée par l'enveloppe de béton armé et les bouchons du bloc pile peut contenir pratiquement tous les effets mécaniques de l'accident et, en limitant la quantité de sodium dispersée dans le bâtiment pile, peut également exclure tout feu de sodium dangereux pour l'enceinte étanche.

Confinement des grands réacteurs à neutrons rapides futurs

Les études faites jusqu'ici conduisent à penser que les conditions d'accidents très pessimistes qui ont servi à définir les enceintes des premiers réacteurs rapides prototypes pourraient être vraisemblablement très adoucies. Il paraît possible de dire dès à présent que l'effet mécanique de toute excursion de puissance réaliste peut être contenu dans une enceinte primaire réduite entourant immédiatement le réacteur, que la possibilité d'un feu de sodium après une telle excursion peut être pratiquement exclue par des dispositions convenables des projets, et enfin que seule une très faible part des produits actifs peut être dispersée dans le bâtiment du réacteur. On peut penser que les grandes centrales à neutrons rapides de l'avenir pourront être installées dans des bâtiments semi-étanches à fuites contrôlées.

ÉTUDES DE SÛRETÉ RELATIVES AUX RÉACTEURS DE RECHERCHE A EAU

Ces études ont été particulièrement développées en France à cause du nombre relativement élevé des réacteurs de ce type et de la politique suivie en vue de l'amélioration de leurs performances. Parmi les piles piscines, TRITON, conçue pour 1 MW, a fonctionné à 2 MW et a effectué des essais jusqu'à 4,6 MW; SILOÉ fonctionne à 15 MW avec expériences en pile. OSIRIS, d'un type mixte piscine-*tank* et à circulation vers le haut, est prévue pour fonctionner à 50 MW; PÉGASE, du type *tank*, a fonctionné à 30 MW et fonctionnera vraisemblablement à une puissance supérieure.

Bien que ces piles soient considérées comme possédant un degré de sûreté élevé et que la cinétique en soit relativement bien connue à la suite des essais des réacteurs SPERT, il a été jugé utile d'améliorer les connaissances dans certains domaines: conséquences des accidents destructifs en ce qui concerne l'intégrité du bâtiment du réacteur, risque de caléfaction, risque d'oscillations de puissance. C'est ainsi qu'ont été entreprises en France des études dans trois directions: études transitoires au moyen du réacteur CABRI; étude des transitoires correspondant aux accidents de refroidissement et des risques de caléfaction au moyen de boucles hors pile; essais de simulation d'accidents de réactivité destructifs au moyen d'explosifs spéciaux

* Seize essais ont été réalisés jusqu'à ce jour.

disposés dans des maquettes représentatives des réacteurs.

Il faut signaler en outre les études effectuées en vue d'une meilleure connaissance des températures du combustible. C'est ainsi que des éléments combustibles équipés de thermocouples de gaine sont systématiquement utilisés au moment des essais d'augmentation de puissance des piles [10].

Réacteur CABRI

Le réacteur CABRI est un réacteur piscine d'un type voisin du réacteur SPERT IV; construit à Cadarache, il est entré en service en avril 1964. Le programme de travail actuellement prévu comprend les étapes suivantes:

1re étape. – Etude des accidents par introduction rapide de réactivité dans une pile à eau. On étudiera l'influence des paramètres suivants: hauteur d'eau, débit, température initiale, puissance initiale, nature du réflecteur. Pendant cette phase seront effectués également:

– des essais d'appareillages de contrôle et de commande de pile et des essais de fusibles;

– les premières études de l'ébullition pendant un transitoire dans une capsule spécialement aménagée pour observer une partie d'élément combustible et permettant d'étudier le mécanisme de limitation de l'excursion de puissance de la pile;

– des expériences en liaison avec les simulations d'excursions de pile par explosifs faites à Toulon, le but étant de déterminer s'il est possible de prolonger par simulation dans le domaine destructif les excursions transitoires des piles.

2e étape. – Dès que la seconde tranche des installations CABRI sera terminée, les problèmes liés aux accidents de refroidissement pouvant survenir dans les installations d'une pile fonctionnant en puissance seront étudiés. Les marges de sécurité par rapport aux risques de caléfaction seront étudiées systématiquement en fonction des caractéristiques de l'élément combustible et du circuit de refroidissement utilisé. On en déduira des critères de sûreté permettant de fixer la puissance maximum des piles piscines.

3e étape. – Cette étape, dont le principe n'est pas encore décidé, utiliserait un cœur composé d'éléments combustibles capables de résister à des périodes très courtes (combustible oxyde vraisemblablement). Une boucle disposée dans une cavité centrale permettrait d'étudier les conditions de destruction de certains types d'éléments combustibles, par exemple de piles à eau ou de piles à neutrons rapides, au cours de différents transitoires représentatifs d'accidents de refroidissement ou de réactivité.

Boucles hors pile

Le Service des transferts thermiques du Centre d'études nucléaires de Grenoble utilise trois boucles hors pile pour les études de sûreté des réacteurs de recherche à eau.

a) La première boucle a servi pour l'étude fondamentale de la convection naturelle de l'eau entre deux plaques verticales en fonction de l'espace entre celles-ci [13]. On a mesuré pour de l'eau dégazée le coefficient d'échange thermique lorsque l'écoulement reste laminaire, ainsi que la valeur des vitesses de l'eau suivant l'intensité du flux de chaleur. Ces résultats ont conduit à des tentatives d'interprétation théorique pour tenir compte de l'établissement du régime et de la modification du profil des vitesses [14].

Actuellement, un canal chauffant constitué par deux plaques collées indépendantes permet la visualisation de l'écoulement; des essais en similitude sont effectués pour déterminer les conditions de refroidissement du réacteur OSIRIS en convection naturelle à la suite d'un incident entraînant la chute des barres. Le cœur du réacteur est surmonté d'une cheminée de 3 m qui se remplit d'eau chaude pendant l'arrêt des pompes. Cette hauteur thermique motrice a été représentée sur la maquette par une charge motrice équivalente d'eau froide variant de 0 à 10 cm. L'écartement des plaques chauffantes est de 2 mm et l'eau utilisée est permutée et saturée d'air dissous. La vitesse de l'eau en fonction du flux superficiel de chaleur est tracée pour différentes hauteurs de la cheminée, ainsi que les valeurs du seuil de flux pour lesquelles on obtient l'expulsion de l'eau du canal *(chugging)*. La mesure des températures de paroi confirme des résultats déjà obtenus [15]: le dégazage intense de l'eau améliore le coefficient d'échange thermique et retarde l'apparition de l'ébullition locale. Au moment du phénomène d'expulsion, l'ébullition locale est amorcée seulement sur quelques centimètres.

b) Une deuxième boucle a été conçue pour schématiser un incident de refroidissement du réacteur PÉGASE. Son but est l'étude des risques de caléfaction encourus par le canal chaud au moment de l'inversion du courant d'eau sous l'effet de la convection naturelle. Le canal d'essai, chauffé par effet Joule, est situé entre deux capacités A et B de 40 l, pressurisées par un vase d'expansion à 7 m de hauteur. Toutes deux sont reliées par un gros tuyau D. On a cherché à représenter le caractère transitoire du phénomène en fixant pour les conditions initiales des valeurs originales:

L'eau située au-dessus du canal en B est froide comme celle qui alimente le cœur du réacteur en écoulement descendant;

L'eau située à la base du canal en A est à une température élevée comme celle qui sort du cœur avant

l'inversion du débit. On choisit comme paramètre la sous-saturation ΔT_{sub} de cette eau contenue dans B;

La puissance thermique est appliquée au canal à l'instant initial sous forme d'un échelon en même temps que l'on ouvre une vanne papillon au sommet du canal pour l'établissement de l'écoulement. On a mesuré le flux de caléfaction en régime transitoire, c'est-à-dire la densité moyenne de puissance qui conduit à la destruction du canal; la longueur des canaux étant fixée à 915 mm et le diamètre hydraulique à 6 mm, on a cherché l'influence de la forme de la section, circulaire ou rectangulaire ainsi que l'influence de la nature du métal, acier inoxydable ou aluminium.

La nature du métal semble ne pas avoir d'influence. Par contre, avec une section rectangulaire de même diamètre hydraulique que la section circulaire, le flux de caléfaction est de 15 à 75% supérieur, suivant la sous-saturation à l'entrée. On doit comparer ces résultats à ceux obtenus en régime permanent. Pendant celui-ci, la situation précédente n'a plus lieu; l'eau surmontant le canal est à la température de saturation; au lieu d'un échelon, la puissance est augmentée graduellement jusqu'à la caléfaction. La flux de caléfaction atteint 1,5 à 2,3 fois celui mesuré en régime transitoire. La visualisation et l'enregistrement des températures montrent que, pour les deux situations envisagées, on a un régime d'écoulement périodique avec expulsion de l'eau du canal. La valeur du flux pour laquelle apparaît cette expulsion *(chugging)* constitue un seuil très marqué qui évolue entre le tiers et les deux tiers du flux de caléfaction pour une pression de 1,75 atm. Bien qu'on ait conclu qu'il n'y a pas de danger pour le réacteur dans le cas envisagé, on note l'influence considérable des conditions extérieures au canal chaud sur la valeur du flux de caléfaction; il semble que le niveau de sous-saturation de l'eau qui surmonte le canal joue un rôle prépondérant.

c) La troisième boucle, dont la construction s'achève, est conçue pour schématiser un incident de rupture de la tuyauterie alimentant le cœur du réacteur PÉGASE. On cherchera à reproduire les conditions initiales de l'incident et les conséquences principales de la rupture; dépression dans le caisson, inversion du débit dans le canal chaud. De plus, le canal chauffant en aluminium aura la même géométrie que l'élément PÉGASE, et on pourra faire baisser rapidement la puissance thermique. Suivant les paramètres choisis, les enregistrements des températures de paroi et des pressions permettront de fixer les limites de caléfaction.

Essais explosifs sur maquettes de piles

Programme des essais

L'objectif de ces essais est d'évaluer les conséquences mécaniques d'accidents de réactivité destructifs se produisant dans une pile piscine. Ces essais doivent permettre de vérifier la bonne tenue de la cuve (risque de vidange de l'eau entraînant la mise à nu du cœur) et la résistance du bâtiment aux efforts dynamiques (gerbe d'eau, surpression, projectiles).

La première tranche d'essais a permis l'étude et la mise au point des charges de simulation et l'exécution de plusieurs essais de maquettes des cuves des piles TRITON et OSIRIS. Les tranches suivantes auront pour but d'étudier les effets mécaniques sur les bâtiments et la tenue des points faibles des cuves (canaux d'expérimentation notamment). On essaiera de relier les effets constatés sur les maquettes aux effets constatés lors des essais destructifs de pile, tels que BORAX I ou SPERT 1. On s'efforcera de mettre au point des charges simulant des excursions à périodes longues (5 à 20 m/s) et d'en comparer les effets à ceux d'excursions de même période effectuées dans CABRI.

Etude et mise au point des charges de simulation

L'excursion maximale connue étant celle de BORAX [16, 17], les caractéristiques de celle-ci ont été choisies comme base de la simulation. Les charges de simulation doivent:

Produire un dégagement d'énergie essentiellement sous forme de rayonnement et de chaleur, en excluant l'énergie mécanique (gaz);

Donner une courbe de puissance en fonction du temps se rapprochant au mieux de la courbe de BORAX I;

Posséder une géométrie la plus voisine possible de la géométrie d'un cœur de pile piscine: forme parallélépipédique, cloisonnement, rapport volume métal/volume eau voisin de 0,6, dégagement de puissance plus important au centre de la charge.

Ce problème a été résolu par la mise au point des éléments suivants:

a) Composition spéciale, ne fournissant pas de gaz, dont la vitesse de détonation est lente (inférieure à 1 000 m/s) et que l'on peut faire varier dans de grandes proportions. Par exemple, pour des tubes détonants avec gaine de plomb, on peut faire varier la vitesse de détonation de 50 m/s à 500 m/s. De même la puissance d'un tube de diamètre initial donné peut varier dans un rapport de 1 à 10 en fonction de la compression;

b) Charges programmées, composées de plusieurs charges élémentaires. Ces charges programmées reproduisent par juxtaposition dans le temps le dégagement d'énergie demandé. Elles comprennent quatre charges élémentaires – sous forme de tubes détonants – donnant chacune un échelon de puissance. L'énergie totale E dégagée par une charge élémentaire est connue par le poids de composition qu'elle contient et le potentiel

de cette composition. Le temps t de détonation de chaque charge élémentaire est mesuré au cours de l'essai. La mesure de ce temps t donne alors la valeur de l'échelon de puissance $W = E/t$.

Il a été vérifié par de nombreux essais que, pour un tube donné, la vitesse de détonation pouvait être considérée comme constante le long du tube. La superposition des quatre marches de puissance est effectuée à l'aide d'un programmeur électronique de mise de feu qui envoie les impulsions de mise de feu des quatre charges élémentaires à des instants prédéterminés. La courbe de puissance effectivement réalisée est restituée après chaque essai grâce à la mesure des temps de détonation et à la connaissance de l'énergie de chaque charge.

Résultats des essais de la première tranche

Ces essais ont été effectués en plein air sur des cuves à échelle 1/10, 2/10, 1/3 et 4/10 reproduisant les conditions des piles piscines, en particulier la hauteur d'eau au-dessus du cœur. Les dégagements d'énergie étaient un peu plus importants que ceux de BORAX I.

A chaque essai, les cuves en béton armé construites spécialement ont été très endommagées ou rendues inutilisables. Par contre, les cuves d'acier mince placées à l'intérieur des cuves en béton n'ont pas été endommagées et l'étanchéité a été conservée.

Les hauteurs maximales des gerbes d'eau ont été importantes, par exemple 6 m pour les essais 2/10, 14 m pour les essais 4/10; ce qui correspondrait à l'échelle 1 à des gerbes de 30 à 40 m.

Les pressions maximales obtenues dans le plan horizontal de la charge sur la paroi des cuves sont de l'ordre de 40 kg/cm^2; les ondes de choc ont des constantes de temps assez grandes (plusieurs millisecondes). Ces valeurs sont nettement supérieures aux valeurs obtenues lors des essais non destructifs de BORAX I et de SPERT I, mais nettement inférieures aux valeurs estimées des pressions lors des essais destructifs de ces piles. Plusieurs raisons peuvent expliquer ces différences: confinement du cœur plus fort dans SPERT I et surtout dans BORAX I, effet d'échelle intervenant dans la dimension de la poche de vapeur. Pour préciser ces points, il est envisagé d'effectuer des essais à échelle proche de l'échelle 1 et reproduisant les conditions de l'excursion destructive de SPERT I.

Les déplacements du sol au voisinage des cuves ont été enregistrés avec des sismographes. Par exemple, lors des essais à l'échelle 2/10, on a mesuré des déplacements verticaux de 10 mm à 1 m et de 0,5 mm à 10 m de la cuve.

L'essai d'un cœur entouré latéralement d'un caisson (maquette de la pile OSIRIS) a montré des résultats voisins de ceux d'un cœur en pleine eau: même hauteur de gerbe, pressions du même ordre, pas de projectiles sortant de la piscine.

ÉTUDES DE SÛRETÉ RELATIVES A L'ÉMISSION ET A LA FILTRATION DES PRODUITS DE FISSION

Les pourcentages de produits de fission émis dans la fusion des combustibles ont fait l'objet d'importantes études, notamment aux Etats-Unis. Le programme en cours de développement en France est orienté plus particulièrement vers l'évaluation des produits de fission émis dans les conditions réelles des réacteurs et vers la mise au point industrielle de filtres à iode.

Des expériences et des travaux préliminaires ont déjà été réalisés dans ces domaines [22, 23] et ont donné lieu à des résultats encourageants. La vérification des filtres à iode par injection d'iode élément radioactif est maintenant effectuée de façon systématique sur les installations du CEA.

CONCLUSION

En conclusion de ce tour d'horizon des études effectuées en France, nous essaierons de dégager les tendances actuelles dans le domaine de la sûreté des réacteurs:

1°) Les accidents de refroidissement constituent, pour la plupart des types de réacteurs, le risque le plus probable et le plus dangereux [19]. L'étude du comportement du combustible au cours des transitoires constitue un objectif important des études de sûreté.

Les accidents de dépressurisation conduisent à des transitoires plus sévères que les avaries des organes de refroidissement. L'amélioration des standards de sécurité des cuves et des circuits sous pression constitue donc un autre objectif important des études de sûreté, afin de réduire encore la probabilité des accidents de dépressurisation. L'intégration des circuits de refroidissement dans un caisson en béton précontraint permet, pour les réacteurs à gaz sous pression, de s'affranchir de ce risque.

Les risques de blocage local du refroidissement doivent être étudiés de près pour les réacteurs à haute puissance spécifique, tels que les réacteurs rapides et les réacteurs à eau. Des dispositifs de détection de l'ébullition devraient être mis au point en vue de prévenir des fusions partielles du combustible.

2°) Les accidents de réactivité sont souvent estimés moins probables que les accidents de refroidissement; mais l'expérience a montré à plusieurs reprises que des erreurs humaines difficilement imaginables pouvaient y conduire. Aussi convient-il de ne pas les négliger, notamment pour les réacteurs à eau et à neutrons rapides, et d'examiner en particulier les conséquences des accidents de réactivité destructifs. Les expériences de BORAX I et de SPERT I sont à cet égard hautement instructives.

3°) Les enceintes de sécurité et les dispositifs de filtration associés doivent être conçus et surveillés de façon à avoir effectivement l'efficacité requise [20]. Les surpressions accidentelles devraient, pour de nombreux types de réacteurs, pouvoir être rapidement réduites par évacuation des gaz à l'extérieur de l'enceinte à travers des filtres.

4°) Les réacteurs futurs, bien que présentant des performances élevées (températures, pressions, puissance spécifique), devraient avoir un degré de sûreté amélioré par rapport aux réacteurs déjà construits, car les résultats des études de sûreté et l'expérience acquise dans l'exploitation des réacteurs auront pu être utilisés pour l'établissement des projets. A la longue, on peut envisager que les centrales nucléaires pourront être installées dans des sites proches des agglomérations, mais ce ne sera sans doute pas avant 10 ou 20 ans, car les incertitudes subsistantes, l'insuffisance d'expérience d'exploitation et surtout l'aspect psychologique tendront à maintenir un certain temps la politique actuelle, qui consiste à installer les réacteurs dans les meilleurs sites disponibles à l'échelle de chaque pays.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/82 France

The concept of nuclear safety in France and influence of safety requirements on reactor design

By F. de Vathaire *et al.*

The paper reviews experience acquired in France in the field of reactor safety.

Since the safety of reactors of the natural-uranium-graphite-gas type is dealt with elsewhere (A/81 and P/81, in this Session), only the safety of other types studied in France is discussed here. They are:

(*a*) Heavy-water-gas reactors;

(*b*) Fast-neutron reactors; and

(*c*) Water-water research reactors of the swimming-pool and tank types.

The safety rules proper to each of the three types are explained, emphasis being laid on their influence on reactor design and on the limits they impose on reactor power.

The corresponding safety studies are described, particular stress being placed on original work developed in these fields. Special mention is made of the experimental facilities built for such studies: the

CABRI reactor (see A/103 and P/103, this Session), in-pile loop for depressurization tests; loops outside the pile; mock-ups etc. Explosive testing on core mock-ups, designed to evaluate the mechanical consequences of destructive power excursions in a swimming-pool reactor, is described in detail.

Finally, safety studies on the emission and filtering of fission products are reported.

A/82 Франция

Проблемы ядерной безопасности во Франции и ее влияние на конструкцию реакторов

Ф. де Ватэр *et al.*

Настоящий доклад представляет собой попытку обобщить опыт, накопленный во Франции по вопросу о безопасности реакторов.

Поскольку газо-графитовые реакторы являются предметом специального сообщения, авторы ограничиваются рассмотрением вопроса безопасности других, известных во Франции реакторов, а именно:

— тяжеловодных реакторов с газообразным теплоносителем,

— реакторов на быстрых нейтронах,

— исследовательских реакторов погружного и корпусного типов.

Рассматриваются проблемы безопасности различных типов реакторов и ее влияние на конструкции реакторов и ограничение мощности.

Представлены материалы исследования проблем ядерной безопасности, в которых особенно отмечаются оригинальные разработки в этой области. В частности, указываются экспериментальные системы, созданные для изучения проблем безопасности: реактор CABRI, петля в реакторе для проведения опытов на сбрасывание давления, внереакторная петля, модели и т. д. Особое внимание уделяется исследованию результатов взрывов моделей реакторов с целью оценки последствий аварий реакторов погружного типа.

Последние разработки проблем безопасности касаются эмиссии и фильтрации продуктов деления.

A/82 Francia

Concepto de la seguridad en Francia e influencia de los requisitos de la misma en el proyecto de reactores

por F. de Vathaire *et al.*

La memoria presenta un resumen de la experiencia adquirida en Francia en materia de seguridad de reactores.

Puesto que los reactores del tipo grafito-gas constituyen el tema de otra memoria (A/81 y P/81), se examina en este lugar la seguridad de los demás tipos de reactores estudiados en Francia:

(*a*) Reactores de agua pesada-gas;

(*b*) Reactores rápidos;

(*c*) Reactores de investigación, de agua, de los tipos « piscina » y « tanque ».

Se estudian los requisitos de seguridad de los diferentes tipos, destacando su influencia en el proyecto de los reactores y las limitaciones de potencia que originan.

Se presentan los estudios de seguridad correspondientes, insistiendo más concretamente en los trabajos originales llevados a cabo en este campo. Se indican especialmente los dispositivos experimentales construídos para estos estudios: el reactor CABRI, circuito experimental en el interior del reactor para ensayos de pérdida de presión, circuitos experimentales fuera del reactor, maquetas, etc. Se mencionan especialmente las pruebas efectuadas con maquetas de reactor cuyo objeto es la evaluación de las consecuencias mecánicas de accidentes de reactividad destructivos que se producen en un reactor de tipo piscina.

Por último se señalan los estudios de seguridad relativos a la emisión y la filtración de productos de fisión.

Development of licensing and inspection of nuclear installations in the United Kingdom

By T. Griffiths and R. Gausden *

The regulation of the nuclear industry in the United Kingdom was outlined in a paper to the IAEA in 1962 [1] in which special reference was made to the Nuclear Installations (Licensing and Insurance) Act, 1959 [2]. The paper indicated the way in which it was intended that the safety provisions of the Act should be administered and the philosophy behind the conditions attached to the licences issued in accordance with the Act. Eight power reactors, all of the gas-cooled, Magnox-clad fuel type have been started up since the date of that paper together with four low power research reactors and a number of other installations subject to the Act. The experience obtained in applying the regulations and conditions to the installations is commented on in this paper and there is an outline of some of the changes made in the initial approach as a result of that experience.

APPLICATION OF THE ACT

The Act requires that any site used other than by a Government Department or the Atomic Energy Authority for the purpose of installing or operating a nuclear reactor shall be licensed, and it gives the Minister the power to prescribe by regulation other kinds of installations concerned with nuclear energy and involving radiation. Regulations [3] were issued shortly after the Act came into force extending its scope by requiring any installation for the manufacture of fuel elements from enriched uranium and for the storage of such fuel elements to be subject to it. The Regulations do not apply, however, to storage incidental to transport or to storage of such fuel elements in connection with research.

As the participation in the field of nuclear energy by industry and non-Governmental research stations developed, it was found necessary to extend the regulations in 1962 [4] to cover installations as follows:

(*a*) For the production of alloys or chemical compounds from enriched uranium, excluding any processes solely for the purpose of chemical or isotopic assay;

(*b*) For the incorporation of enriched uranium in test rigs designed for irradiation in a nuclear reactor;

(*c*) Subcritical assemblies using enriched uranium;

(*d*) For the storage or processing of irradiated nuclear fuel other than storage incidental to transport.

The intention in making these Regulations was to ensure that installations handling enriched uranium in such quantities that any risk of criticality could arise or handling irradiated nuclear fuel should be subject to the Act. The classes of installations described in the two sets of Regulations embrace all of the existing relevant installations outside Government departments and the Atomic Energy Authority. It is intended that research and development should not be subjected unnecessarily to the special control and legal regime imposed by the Act. The position is therefore under continuous examination with a view to defining the optimum scope of the Act at each stage in the development of the industry.

ORGANISATION OF THE INSPECTORATE

The Act makes provision for an Inspectorate to be appointed to assist in its execution and this was formed and organised in the following sections:

(*a*) Assessment Section for Power Reactors;

(*b*) Assessment Section for Research Reactors and other installations;

(*c*) Section providing on-site inspection;

(*d*) Technical Section;

(*e*) Standards Section.

It was the original intention that the Assessment Section would be responsible for the design safety assessment and all safety conditions formally imposed on the licensee during commissioning and operation and that the Site Inspection Section should have duties limited to checking for compliance with the formal conditions, with no discretion for modifying the conditions other than by referring them back to the Assessment Section. This arrangement has, however, been found to be cumbersome and not in the best interests of safety, as conditions formulated to meet certain circumstances have been found to be inappropriate

* Inspectorate of Nuclear Installations, Ministry of Power, London.

for other circumstances not foreseen at the time. To anyone who has not been concerned with framing the conditions in the first instance, the original circumstances may not be known and it is necessary for every course of action to be examined for its inherent safety to ensure that compliance with a condition is not the only criterion. In order to remedy the situation, the Assessment Section for power reactors now deals with engineering and scientific matters from the earliest design stage up to a point at which nuclear fuel is loaded, and an Operational Safety Section has been formed charged with approval of commissioning procedures and operating rules and also with site supervision and compliance. It might be expected that under such an arrangement an inspector responsible for a site might pay too much attention to some aspects of the plant to the detriment of a detached systematic and routine examination of other important matters, but it has been considered that this can be guarded against by an effective system of supervision.

For the research reactors and types of nuclear plant other than reactors, the Research Reactor Assessment Section has been given full responsibility for all aspects of the installation from the first design assessments to formulation of conditions and rules and to site inspection for compliance with the conditions and rules.

From time to time in the operation of the plant situations arise, often due to external influences, such as transients on the power network, which can have unexpected effects on the operation of the plant. Such effects can be very revealing, particularly with regard to the performance of systems; a close examination of the occurrences and their consequences justifies close study. For this reason a Special Projects Section has been formed to give detailed attention to such matters. The Special Projects Section is also charged with progressing and encouraging the development of emergency schemes and the training of personnel to meet accident conditions on the sites. This section now replaces the Standards Section which has been coalesced with the Technical Section.

The organisation of the Inspectorate is now divided into the following sections:

(*a*) Power Reactor Safety Assessment;
(*b*) Special Projects;
(*c*) Power Reactor Operational Safety;
(*d*) Research Reactor Assessment;
(*e*) Technical.

POLICY WITH REGARD TO SITES

Seven sites had been selected for power reactors prior to the coming into force of the Act and these have all been examined with respect to the population density and distribution in the neighbourhood. A system of weighting factors to allow for distance from the reactor of the population has been employed, as set out in Farmer's paper [5]. All seven of the sites were found to have a site rating factor not exceeding 750 and to lie within the Class 1 defined in Farmer's paper. Gas-cooled Magnox reactors with steel pressure vessels and external cooling ducts on Class 1 sites have been taken as the basis for assessing comparatively other sites and other reactor types or designs.

In 1962, it was accepted that a prestressed concrete reactor pressure vessel might be employed in place of the steel vessel on a Class 1 site for the same reactor system and with the same rating limitations. In determining the limits which are placed on the operating power of the Magnox reactors, maximum fuel element temperatures and ratings and maximum graphite moderator temperatures are determined on the basis of a postulated accident, which is taken to be a depressurisation of the main pressure circuit arising from the rupture of an external gas cooling duct. The maximum ratings are determined on the basis that there must be a low probability of such an accident leading to a channel fire and the accident is usually thought of in terms of a burst bottom duct as it can be shown that temporary stagnation of gas flow within the core can be occasioned by a rupture of a critical size at that point. Further consideration has been given to the prestressed concrete pressure vessel which in the design put forward contains the heat exchanger system and is without external cooling gas ducts. No failure of the vessel can be conceived which would lead to a core incident or invalidate the assumption that the containment of fission products arising from a core incident can be controlled. It has therefore been accepted that such reactors may be constructed on a site which falls within the Class 2 category as determined by the site rating factors. It is no longer relevant to postulate an accident in terms of a ruptured main gas duct, but no case has yet been put forward or agreement given to the operation of the integral prestressed concrete pressure vessel reactor at ratings higher than those previously set for steel pressure vessels with external ducts.

SUBMISSION FOR LICENCE

With the material departure in the design of the reactor systems from the Calder Hall design, it is now necessary for the initial application for a licence to include information on fundamental safety aspects of the proposed system. Information must be given regarding the main pressure containment system and cooling arrangements, together with an advance layout of the site showing expected radiation contours and arrangements to be made on the site for dealing with active effluents and the handling of irradiated fuel. Details of the control and shut-down system proposed are necessary and the submission has to include a

summary of the design assumptions affecting safety of the system and an indication of the fault studies and other investigations and tests planned to support those assumptions.

CONDITIONS ATTACHED TO LICENCES

The conditions which bear on safety and which have to be attached to the licence provide the necessary legal basis upon which the work of the Inspectorate is founded.

An attempt has been made to keep the conditions as brief and clear as possible and as free from technical data which might very well have to be amended fairly frequently. Nevertheless, as might be expected, the expression of general ideas and requirements in words has led to problems of interpretation and of situations arising where in unforeseen circumstances the wording can be at variance with what is sensible.

It has been the practice to attach conditions to the licence applicable during constructional work on the site covering only restrictions on the disposition of the site, the approval of plans, designs, or specifications and requirements for the examination, testing and reporting of failures and defects in plant under construction. In order to ensure, however, that construction work has not been too far advanced before the design details have been made available to the Inspectorate, the conditions have recently been strengthened to provide that the construction of parts of the plant shall not proceed until adequate information has been made available.

On completion of construction and before approval is given to load fuel, a variation to the licence conditions is issued embracing additional requirements appropriate to the safety of the operation of the reactor. These conditions allow for preservation of records, fencing, duties of authorised persons, health physics requirements, and stipulate the arrangement which must be made for ensuring that the plant is operated safely in accordance with written instructions. Conditions also require the licensee to make preparations for dealing with any accident or emergency on the site.

The basis of a licence is the site, as has previously been stated, and for this reason it is necessary for the perimeter of the site to be clearly defined in the licence documents and to be suitably marked on the ground. From the legal point of view this defines the area where security of tenure must be maintained and the boundary within which the obligations of the licensee are imposed. The protection of members of the public who live in the neighbourhood of the installation is safeguarded by the examination of the location of the plant with respect to houses before the licence is issued, by the control of development in the neighbourhood of the site after the licence has been issued, and by the effective shielding of the plant to ensure that the radiation contours declared when the application was made for a licence are met. It has been found that in general, provided the radiation contours on the licensed site in areas not subject to special restriction are sufficiently low to meet the irradiation dose limitations for workers on the site, other than classified workers, then this ensures that passers-by are sufficiently safeguarded.

The intentions behind all requirements for health physics safeguards are based on the recommendations of the International Commission on Radiological Protection as interpreted by the Medical Research Council [6] and there are a number of licence conditions bearing on health physics precautions and the medical supervision of classified persons. It has been found possible to define closely the procedures to be followed in obtaining evidence of direct radiation doses by film badges and dose meters. Air-borne radioactivity concentrations are controlled for most part by ensuring that the assessment of the plant design ensures adequate provisions together with good operator discipline. From time to time, however, operators are exposed to concentrations of radioactive particles; an effort is made to quantify such exposure in terms of radiation doses so that records of an individual's exposure shall be sufficiently embracing. This has been found a difficult matter to express in regulations and the relevant conditions have had to be extended. Even then reference has had to be made in the conditions to the ICRP recommendations.

The Nuclear Installations (Licensing and Insurance) Act, 1959, gives the necessary power for conditions to be imposed on the licensee with respect to the handling, treatment and disposal of any nuclear fuel irradiated on the site and conditions are in force bearing on the keeping and storage of fuel on the site, containers for the transport of irradiated nuclear fuel and for the disposal of the nuclear fuel. Documents forming part of the approval for the containers for nuclear fuel form the basis of control to ensure that:

(*a*) A flask of approved construction and inspection is used;

(*b*) The contents fall within acceptable limits of radioactivity, etc.;

(*c*) The limits for direct radiation are met when the flask is secured;

(*d*) The surface contamination value specified is not exceeded;

(*e*) The maximum of external surface temperature is not exceeded;

(*f*) Where relevant, police and highway authorities are advised of any movement off the site in advance.

In the case of the Magnox-clad, natural uranium fuel from the nuclear power stations, the criterion for the contents is based on the maximum permissible potential release of activity not exceeding 3 curies of krypton and 2 curies of other volatile fission products.

An obligation is also placed on the licensee to ensure that at all times there is an organisation to deal with any accident involving the nuclear fuel in transit. This organisation may be part of the licensee's own establishments or, by arrangement, be provided by other licensees or the UKAEA if placed geographically more advantageously.

DESIGN SAFETY ASSESSMENT

Design and manufacture of components for the first power reactors in the nuclear power programme were complete before the Nuclear Installations (Licensing and Insurance) Act, 1959, came into force, while work on a number of later reactors was well advanced, particularly the main pressure vessel and gas circuits. The first safety assessment of the design produced by the licensee, called the Intermediate Safety Report (see *Guide to Safety Report Requirements* [7]), was provided at a stage of the construction where objections to any major aspect of the design would have been costly had such a situation arisen.

For new plants, discussions are under way with the licensee — in this case, the Central Electricity Generating Board — with a view to the replacement of the Intermediate Safety Report referred to previously, which normally cannot be completed until some two years after construction commences, by an alternative arrangement. It is the practice, of course, for much of the detailed design work to be done after the licence has been issued and construction work has started. The licensee is asked to supply a programme chart for this work indicating the dates by which the various designs must be frozen if constructional target dates are to be met. This enables the Inspectorate to indicate what design aspects are to be the subject of investigation and to say on what dates the design information must be made available. For the gas-cooled reactors, construction of the main pressure vessel is normally planned to commence some six months after the licence is issued and so the main design information must be cleared within a few months of the licence being issued. The remaining design information is then provided in sequence. A final safety report summarizing the whole position is still required and this must be provided about six months before the fuel loading of the reactor is planned to be commenced.

The Inspectorate has sought to influence the situation where the design of the plant has tended to proceed without due regard to the contribution which the operator must make to safety. The effort to make the plant safe without regard to any operator action carried to its logical conclusion could well give rise to a complexity of design providing safety problems on its own account. Encouragement is being given for the safety reports to take account of the contribution of the operator to safety and to indicate what safeguards have been left in the hands of the operator, where the alternative is complex or otherwise unsatisfactory. If such a proposal is approved the safety report will then ensure that the operator can be made clearly aware of his responsibility for specific aspects of plant operation in addition to his general responsibility. Where such direct reliance is placed on the operator, it is vital, of course, that the layout of controls and the availability of instrumentation are carefully considered to facilitate prompt warning and action by the operator.

INSPECTION DURING CONSTRUCTION

For steel pressure vessels it has been possible to make use of the experience expertise and efforts of the independent Insurance Societies to provide all direct supervision of materials, construction and testing. For prestressed concrete pressure vessels, it was thought that in order to ensure that a comprehensive inspection was carried out it would be necessary to have a similar system to that used for the steel pressure vessels, namely, one agent in the form of an Insurance Society responsible for inspecting the whole of the pressure vessel construction. In this way there was little chance of any overlapping of inspection duties and, more important, no chance of certain sections having a cursory inspection, or even at worst none at all.

No one agent, however, had the necessary breadth of experience to be competent to handle the whole task and the Inspectorate eventually accepted that the responsibilities be divided. The Insurance Society would carry out the same function as before and would be responsible for the detailed inspection procedures on all steel parts of the vessel except for reinforcing and stressing ligaments, whilst a firm of consulting engineers would cover the reinforced concrete construction and be responsible for all the stressing arrangements. There was still however the possibility with this divided arrangement that the line of demarcation was not sufficiently clear cut to ensure that all parts of the plant were inspected, and a liaison and co-ordinating engineer would be necessary. One of his main tasks would be to arrange frequent discussions between the inspecting bodies so that they may become aware of each other's day-to-day problems and be able to build up an expertise in this type of construction.

In practice, the purely civil engineering aspects of the work has been relatively straightforward so far, and the liaison maintained on site has been good; there is a clear indication that the co-ordinating engineer performs an essential function. He should always be a senior engineer with direct responsibilities.

COMMISSIONING OF THE REACTOR

The licensee is expected to submit a complete programme of tests for the commissioning of the reac-

tor and a complete set of test documents. The papers for each test planned are made up under three headings:

(*a*) Purpose of test and prognostication of results;

(*b*) The minimum safety provisions required to be in force during test;

(*c*) Details of test procedures.

(*a*), together with the programme, enables a check to be made that in so far as it is possible every assumption of safety considered in the safety assessment is demonstrated while the prognostication of the results provides figures against which the final results can be judged. (*b*) makes quite clear what aspects of the safety circuits or other safeguards are essential for the duration of each test and these can be checked independently. This has been found necessary as very often additional safety equipment is in operation during the test which can be disconnected without prejudice to immediate safety if trouble is experienced of if modifications are to be made. Unless all concerned are aware what safety equipment can be considered to be redundant in the immediate situation, commissioning may be unnecessarily delayed, or there may be the chance that essential safeguards are circumvented. The vast bulk of the documents formed by section (*c*) are available to be studied if further detail of a test becomes necessary or if special attention is to be paid to certain tests.

OPERATION OF THE REACTOR

The documents providing operating limitations and controls for power reactors as provided for by a condition of the licence, are the operating rules and approved drawings.

Only minor difficulties have arisen in conjunction with the operating rules, which are subject to formal approval. Where operating limits are expressed in figures, experience has shown the necessity for ensuring that if the limits are not directly shown by instruments then an appendix to the rules must define exactly how such limits shall be calculated or derived from observed readings, and the importance of saying precisely whether the limits are to be interpreted as a ceiling or an average and if so how this is to be interpreted.

A licence condition provides for drawings of any part of the plant to be approved formally, and when this has been done, changes to the plant or systems shown on the drawings may not be made without formal consent. The drawings which have been approved have not always been drawings on which matters solely important to safety have been shown; many of the drawings are general assembly drawings or schematics. Where this has been the case consents have had to be issued to cover changes to equipment shown on the approved drawings although the change in question may have no safety importance. Consideration is being given to methods whereby minor changes to plant not affecting safety can be made without invoking the formal requirement for approval or consent.

For the small and less complex research reactors, special drawings are required in defining safety aspects of the plant and these have been incorporated in the general specification required before plant operation is authorised which cannot be changed without formal approval.

Experience has shown that a most important contribution made to safety by this system of approved drawings is that at the time the reactor commences operating at power, and when all the minor changes found necessary in the early commissioning stages have been made, an up-to-date record is available of the plant and systems. Unreliable drawings may well provide a misleading starting point for consideration of subsequent changes, with the possibility of an unsafe modification being initiated.

DANGEROUS OCCURRENCES AND OTHER REPORTS FROM LICENSEE

The Nuclear Installations (Licensing and Insurance) Act, 1959, requires a licensee to report any dangerous occurrence of a kind prescribed by regulations. Regulations [8] have been issued specifying a number of kinds of occurrences and although no dangerous occurrence within the ordinary meaning of the term has occurred, nevertheless a number of incidents which appeared to be covered by the regulations have been reported. A full examination of the circumstances in each case has been made and it has been concluded that they were not in fact within the terms of the regulations. It had thus become clear that certain parts of the regulations were difficult to interpret and might be held to cover occurrences which were in no way dangerous.

To meet this difficulty, amendments [9] have now been made to the regulations by deleting two provisions. These concern the reporting of explosions or outbreaks of fire in any place where there is radioactive or toxic material and the reporting of occurrences leading to the shutting down of the plant and which prevent resumption of operation within one hour. Parts of the regulations, however, still effectively cover such happenings, if indeed they are dangerous, and they will still be reportable in terms of these parts.

The normal channel for information to the Inspectorate concerning the routine operation of the plant, is through the Site Inspector and his reports. By this means matters of importance can be distinguished at the site, and the amount of paperwork which needs consideration at headquarters' level reduced to the minimum. The organisation of the Operational Safety Section takes this requirement into account. However, a licence condition requires notification to the Minister

to be made where a single dose of whole body radiation has exceeded 12 rems, or if contamination on a person has continued for three days, of if an "undue" amount of activity has been ingested or inhaled. "Undue" is interpreted by reference to ICRP maximum permissible concentrations and the length of time they were experienced. The notification ensures that an Inspector can be directed to the site to investigate the situation without delay.

RECORDS

A very important and positive aspect of the responsibility of the Inspectorate is seen to be the compilation of records, and to ensure that this is done in such a way as to serve two main purposes.

The first purpose relates to the liability provision of the Act, whereby claims on account of injury may be presented within a period of thirty years. This far exceeds periods of liability applied to any other industry and human memory is likely to be of little service. Records are kept by the licensee of individual health physics measurements for this period of time, but it is considered to be very important that any unusual features of these records should be examined independently as soon as possible, and a clear statement of the situation placed on record. It is found that such enquiries can seldom be confined to an examination of health physics records and interviews with employees. A knowledge of the operating conditions or circumstances of the plant is essential.

The second purpose concerns the necessity for extracting on a systematic basis all operating and engineering experience which might bear on the adequacy or otherwise of preconceived ideas on safety principles, and to build up a firm basis upon which a regulatory body can operate.

As mentioned in the previous section, it is considered that at this stage of development of the industry, formal reports from the licensee are not a satisfactory way of obtaining information required for records. Reliance must be placed on the efforts of properly trained safety officers in close contact with the installations.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/181 Royaume-Uni

Règles concernant l'octroi des autorisations d'exploitation et l'inspection des installations nucléaires au Royaume-Uni

par T. Griffiths et R. Gausden

Le contrôle des installations nucléaires importantes, à l'exclusion de celles exploitées par l'United Kingdom Atomic Energy Authority, est réglementé par la loi de de 1959 sur les installations nucléaires (autorisation d'exploitation et assurance) *(Nuclear Installations (Licensing and Insurance) Act)*. La loi exige que l'exploitant d'une installation nucléaire ait une autorisation d'exploitation et définit la responsabilité du titulaire de l'autorisation et les dispositions qu'il doit prendre pour couvrir cette responsabilité. Il est exigé que l'autorisation d'exploitation soit assortie des conditions qui peuvent être nécessaires ou désirables dans l'intérêt de la sécurité.

La loi spécifie qu'aucun emplacement ne peut être utilisé pour l'installation ou l'exploitation d'un réacteur nucléaire sans une autorisation d'exploitation et donne pouvoir de réglementer concernant d'autres types d'installation nucléaire selon une large définition.

L'application de la loi exige qu'il soit déterminé quelles installations doivent être réglementées et de quelles conditions chaque autorisation d'exploitation doit-elle être assortie dans l'intérêt de la sécurité.

Le mémoire traite des principes qui ont été appliqués et indique comment ces principes ont été suivis.

Des commentaires sont donnés sur l'expérience des quelques premières années de l'application des règlements et des conditions et on indique certains des changements qui furent effectués à la suite de cette expérience. On traite également du développement de la politique suivie pour l'emplacement des réacteurs nucléaires et de l'organisation de l'inspectorat qui a été mis sur pied pour l'application de cette loi.

A/181 Соединенное Королевство

Разработка системы лицензирования и инспекции для ядерных установок в Великобритании

Т. Гриффитс, Р. Гозден

Контроль за основными ядерными установками, исключая те, которые эксплуатируются Управлением по атомной энергии Великобритании, осуществляется в соответствии с законом о ядерных установках (лицензирование и гарантии) от 1959 года. Этот закон требует наличия лицензии у лица, эксплуатирующего ядерную установку, и определяет обязанности и условия, возлагаемые на обладателя лицензии. Это — требование, которое в докладе будет отнесено к условиям получения лицензий как необходимое или реальное в интересах безопасности.

Закон гласит, что никакая площадка не может быть использована для установки или эксплуатации ядерного реактора без лицензии, и обеспечивает предписания для установления правил сооружения и использования других типов ядерных установок в широких пределах.

Для соблюдения этого закона необходимо определить, какие ядерные установки могут сооружаться и какие условия в интересах безопасности должны ставиться при выдаче каждой лицензии.

В докладе описываются применяемые принципы и указывается, как они воплощались при первоначальной разработке правил и условий выдачи лицензий. Рассматривается опыт первых нескольких лет по применению этих правил и условий, и описываются в общих чертах те изменения, которые произошли в результате накопления этого опыта. В докладе говорится также о разработке критериев выбора площадки для ядерных реакторов и о развитии инспекционной организации, созданной для контроля за выполнением вышеуказанного закона.

A/181 Reino Unido

Evolución de las normas de autorización e inspección de instalaciones nucleares en el Reino Unido

por T. Griffiths y R. Gausden

El control de las instalaciones nucleares importantes, excluyendo las que son explotadas por la Atomic Energy Authority del Reino Unido, está regulado por la Ley de instalaciones nucleares (autorización y seguros) de 1959. Esa ley estipula que el explotador de una instalación nuclear necesita una licencia, y establece la responsabilidad de quien obtiene esa licencia y las medidas a tomar para cubrirla. Es preceptivo que se incluyan en la licencia cuantas condiciones sean necesarias o aconsejables en interés de la seguridad.

La ley especifica que un emplazamiento determinado no podrá ser utilizado para instalar o explotar un reactor nuclear sin la debida licencia y autoriza para que se formule la reglamentación correspondiente a otros tipos de instalaciones nucleares que quedan englobadas en una amplia definición.

La aplicación de la ley requiere la determinación del tipo de instalaciones a regular y las condiciones que deberán incluirse en cada licencia en atención a la seguridad.

La memoria trata de los principios que se han aplicado e indica cómo se estipulaban en el texto original de la reglamentación y las condiciones. Se hacen comentarios sobre la experiencia en estos primeros años de la aplicación de las condiciones y normas regulativas, describiéndose las modificaciones efectuadas como consecuencia de esa experiencia. La memoria hace también referencia a la evolución de la política sobre emplazamientos de reactores nucleares y a la organización del cuerpo de inspectores, fundado para dar cumplimiento a la Ley.

The safety of UK gas-cooled reactors - a review

By G. R. Bainbridge,* L. F. Bowles, L. Cave,*****
J. R. Harrison ** and J. M. Yellowlees *******

The 1958 Geneva Conference received reports on the safety aspects of the Calder Hall uranium-Magnox, CO_2-cooled, graphite-moderated reactors [1] and larger reactors of that type optimised for electricity generation [2].

The eight Calder and Chapelcross reactors have been operating for over 45 reactor years. Since 1962 power reactors at Bradwell, Berkeley, Latina, Hunterston and Trawsfynydd have been set to work [3].

Because of the possible serious consequences of an accident at one of these large reactors many studies have been done to confirm and improve their safety. Examples are given here of improvements arising from these analyses and from operating experience.

At Windscale the experimental advanced gas-cooled reactor [4], using UO_2 fuel in stainless steel cans was commissioned in 1962. Commercial AGRs may merit a relaxation of site requirements in the UK, as may reactors having special features such as integral pressure vessels [5]. The safety of these designs is discussed in this paper.

SYSTEM CHARACTERISTICS

Magnox reactors have basic characteristics which, singly or together, affect reactor safety. These are summarised in the following:

To prevent unacceptable deformation of the heat transfer surface both the maximum can temperature and the aerodynamic forces, and hence the channel coolant flow, must be restricted. The former is in the region of 500 °C. These therefore set a limit to the maximum fuel rating, which is also affected by the requirement that the uranium α-β phase transition temperature (665 °C) should rarely be exceeded. There is an incentive to increase specific rating within these limits by redesign of the fuel element [6]. Such redesign might double maximum ratings which are now about 6 MW/t.

As the natural uranium fuel requires a lattice pitch of 20–30 cm, the core power density is relatively low (less than 1 kW/litre) and the large mass of graphite can act as a heat sink following a fault.

Low core power density, coupled with high coolant pressure, may for a large reactor lead to pressure vessel dimensions which introduce construction problems. In practice, steel vessel working pressures have been kept below 300 lb/in^2 until further experience is obtained of construction techniques. Also the higher the gas pressure the greater the graphite-CO_2 reaction rate, so even with concrete vessels coolant pressures of 400 lb/in^2 are not being exceeded.

Because the overall temperature coefficient of reactivity becomes positive with irradiation the system is not self-stabilising and control rod movement is required if core temperatures are disturbed. However the low rating and power density lead to relatively slow changes for most faults. Thus, although some fuel cans operate only 150 °C below their melting temperature there is adequate time for automatic or manual protection.

The reactors are not inherently protected against all accidents and for severe faults an automatic trip system is necessary. The designer, seeking a balance between safety and economy, may specify the performance of the protective system so that calculated fault behaviour is acceptable for a given fuel rating. Thus at first sight it might appear that the fuel rating is limited on safety grounds, but this is not necessarily so.

Base-load operation and on-load refuelling reduce to some extent the need for a rapid control rod withdrawal procedure at start-up. Hence the magnitude of the possible transients due to incorrect rod movement is less than for many other systems.

Three characteristics which have required design attention to ensure safety are:

(*a*) Accumulation during irradiation of stored energy in graphite;

(*b*) Increase during irradiation of chemical reactivity for the graphite reaction with air; and

(*c*) Near coincidence of melting and ignition temperatures for Magnox at about 645 °C.

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** Atomic Power Projects, Leicester.
*** United Power Company, London.
**** Central Electricity Generating Board, London.
***** The Nuclear Power Group, Knutsford.

The graphite working temperature in regions of high neutron flux may be selected so that stored energy does not markedly influence fault behaviour. The significance of the graphite-air reaction following depressurisation has been studied and methods for control exist. Faults severe enough to cause general melting of Magnox can be eliminated by design and damage to individual channels made extremely unlikely.

These problems are taken into account in the following fault sections.

MINOR FAULTS

Table 1 shows a selection of the faults other than loss of coolant considered for a typical, steel pressure vessel, Magnox reactor. Fault analyses have been done to determine specifications for:

(*a*) Reliability of plant;

(*b*) Protection equipment type, arrangement, performance and response; and

(*c*) Operating rules.

Protection may be manual, or if prompt action is required, automatic. In the latter case at least two independent sets of measurements, with trip initiation by either set, are used to ensure that the fuel is undamaged. The problems of following this principle for some reactivity faults are detailed in another paper [7]. Unless faulty fuel is already in the reactor, release of fission products from cans will not be caused by these minor faults. A restriction in a single channel leading to a flow reduction of more than 40% is necessary before the hottest can there will melt.

Warning of possible restriction must therefore be obtained by routine observation of burst cartridge detection equipment and items such as:

(*a*) Coolant filters for excessive accumulation of debris;

(*b*) Internal components such as insulation for possible displacement;

(*c*) Graphite channel walls for spalling;

(*d*) Fuel for excessive distortions; and

(*e*) Malfunctioning of fuel handling equipment.

Nevertheless some local deterioration might go undetected, to the extent that a fuel element reaches excessive temperature during normal operation or in a fault; the implications of ignition of fuel in a channel have been studied.

Molten material may fall past the fuel support and gag but penetration or weakening of the pressure circuit is prevented by fairing plates beneath the core. The pressure circuit therefore remains an effective containment.

Figure 1 shows a typical estimated temperature distribution between a burning channel and one adjacent. Whether the reactor is at full power or shut down,

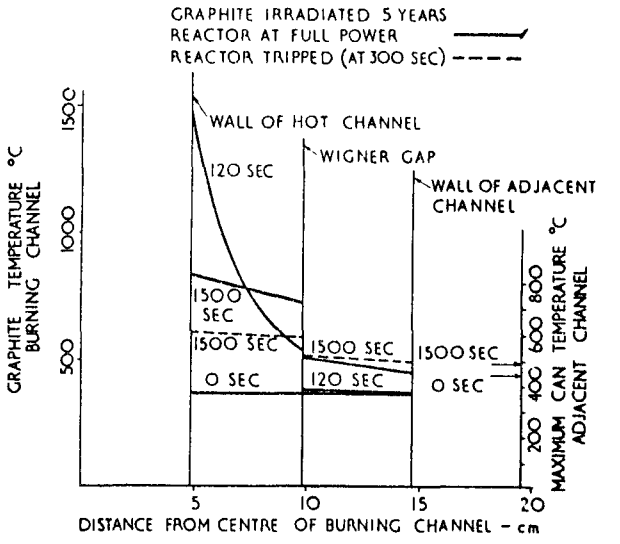


Figure 1. Temperature profiles following single channel fire

fuel temperatures in the adjacent channels change only slightly and remarkably slowly. In none of the cases studied with a wide range of data has there been any indication of dangerous overheating in adjacent channels.

Fission products entering the coolant in unacceptable quantity can be removed if necessary after shutting down by discharging the coolant to a treatment plant permanently connected to each reactor, so avoiding significant leakage.

DEPRESSURISATION FAULT—NONINTEGRAL STEEL VESSELS

Experience in many countries has shown that steel pressure vessels built to high standards (e.g. to the ASME or BSI 1 500 Codes) are unlikely to fail in conventional usage. In nuclear applications the effect of irradiation and difficulties of inspection in service are adverse factors; but by working to an even higher standard and by monitoring for temperature and irradiation effects, it is considered that the risk of catastrophic failure in service is made extremely small.

As an example of a large, but not catastrophic, pressure circuit failure the fracture of a duct in a nonintegral steel circuit has been studied in detail. Protection against such failures has been arranged without upsetting the balance between economy and safety. Such protection is an accepted design requirement for reactors in the UK.

The transient forces on essential equipment * during depressurisation are calculated using where available methods verified experimentally [8]. The designs ensure

* Fuel and flattening; control rods, their guide tubes and gags; graphite layers and bricks; core charge pans, side restraints, gas seals and diagrid; blowers; main gas valves, ducts and bellows; biological shield.

Table 1. Typical minor faults

Plant	Fault	Prevention	Detection	Effect	Remarks
Reactivity					
1 All coarse rods	Uncontrolled withdrawal	Generally gravity fail-safe Self checking control circuits	Outlet gas temperature, fuel temperature/flux	Sector rod correction Operator correction reactor trip	6 minutes with sector correction to trip 2 minutes to trip
2[a] All coarse rods	Uncontrolled withdrawal	As for 1	Doubling time/flux/fuel temperature (decreased trip-level)	Operator correction reactor trip	2 minutes to trip
3 All sector rods [c]	Uncontrolled withdrawal	At least 9 independent groups As for 1	As for 1	As for 1	1 minute to trip
4[a] All sector rods	Uncontrolled withdrawal	High integrity interlocks As for 3	As for 2	As for 2	1 minute to trip Reactivity rates up to 2 MN/s
5 One sector group of rods	Uncontrolled withdrawal	As for 1	Outlet gas temperature Fuel temperature	As for 1	5 to 15 minutes to trip Temperature protection spatially distributed
[Faults not included above: Misalignment of control rods (axially): Maldistribution of control rods (radially): Mismanagement of absorber/fuel changing programmes]					
Core coolant reduction					
6 (*a*) One blower	Failure, mechanical or electrical	High reliability supplies and controls	Outlet gas temperature Blower state indication	Sector rod correction Operator correction	Less than 25% core flow decrease
(*b*) One main gas valve			Fuel temperature		
(*c*) Core-pressure vessel gas seals		Double 100% duty seals Small seal radial area			
7 (*a*) Two or more blowers	Failure, mechanical or electrical	As for 6(*a*) and 6(*b*)	Outlet gas temperature	Sector rod partial correction	Less than 50% core flow decrease for 2 blowers or valves
(*b*) Two main gas valves			Blower state indication Fuel temperature	Operator correction/reactor trip	Natural circulation and/or pony motors will remove after-heat
[Faults not included above: Closure of more than 2 valves (inconceivable): Fault in coolant flow auto-control (limited control range)]					
Turbines and boilers					
8 Turbine	Trip or fail		Inlet gas temperature Fuel temperature	Sector rod correction Operator correction/reactor trip	Safety valves limit steam pressure rise
9[b] Steam pipe	Break (*a*) To CO_2 coolant		Water in CO_2 If major – safety valves	Operator correction/reduce reactor power Steam/CO_2 discharged	Tube to be isolated Partial discharge only
	(*b*) To atmosphere		Inlet gas temperature Fuel temperature	Operator correction/reactor trip	
10 Feedwater pipe	Break		Inlet gas temperature	Operator correction/reactor trip	
Feedwater controls	Error		Fuel temperature		
Feedpumps	Failure, mechanical or electrical	Standby pumps (100%) and emergency supplies			Reactor tripped if due to power failure
[Faults not included above: Boiler isolating valve closure (less effect than turbine trip)]					

[a] Less than 5% initial reactor power, all other cases greater.
[b] Only in 9 (*a*) is any of the coolant released to atmosphere; in no case are fission products likely to enter the coolant.
[c] Automatic rod groups independently controlling sectors or regions of the core.

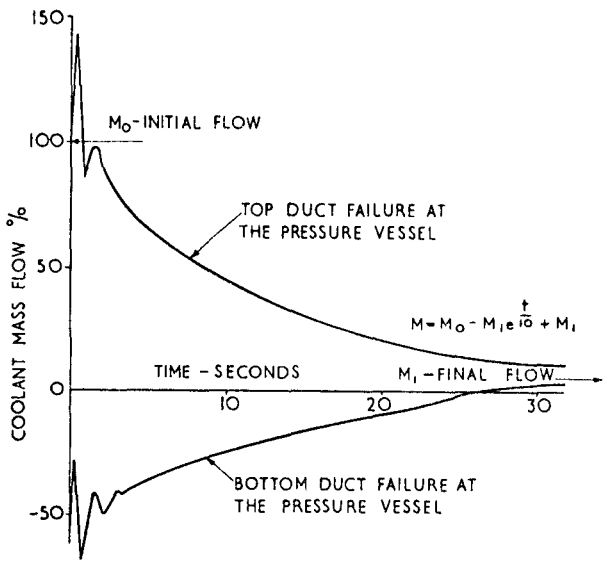


Figure 2. Core coolant flow during depressurisation

a rapid reactor shut-down and adequate core cooling. It has been checked that the containment provided by the pressure circuits could be re-established by emergency procedures and that the biological shield could form an additional containment system.

Core flow variation with time for complete severance of a top or bottom duct may be seen for one design in Fig. 2. All but one of the main blowers were assumed to continue at full speed during the transient period. The nominal depressurisation time constant is 10 seconds. For either failure core cooling is adequate

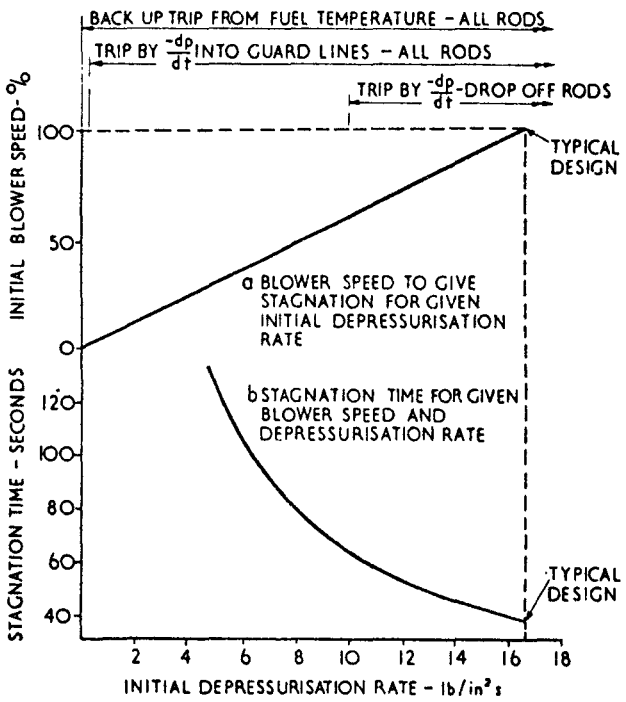


Figure 3. Blower speeds for core flow stagnation and related stagnation times over a range of depressurisation rates

during the initial transient, with 150% forward or 50% reverse flow respectively, and the fuel temperature rises are comparatively slight.

Calculations show that flow stagnation, leading to a severe temperature rise in tens of seconds, could occur with a smaller tear of a bottom duct.

Figure 3(*a*) shows the initial depressurisation rate which could cause stagnation at different initial blower speeds. In Fig. 3(*b*) the related duration of stagnation may be read off.

Can temperatures, estimated for the duration of coolant stagnation and immediately afterwards, depend on several parameters, namely:

(*a*) Initial fuel ratings and temperatures;

(*b*) Stagnation time and subsequent coolant flow;

(*c*) Reactor shut-down characteristics;

(*d*) Magnitude and time variation of after-heat;

(*e*) Thermal capacity of fuel and surrounding graphite; and

(*f*) Radiated heat-transfer characteristics.

Temporary operational variations of the parameters are considered, e.g. adjacent empty fuel channels or other causes for local rating increases.

Figure 4(*a*) illustrates the effect of amount and time of absorber insertion, while Fig. 4(*b*) shows the effect of stagnation duration. A depressurisation rate exceeding about 0.1 lb/in² s would cause all the control rods to run into the core normally in 5 to 15 s. Near 10 lb/in² s depressurisation rate devices attached to some of the rods ("drop-off" rods together worth 3% reactivity) would unlatch them from their suspensions to enter the core even sooner in about 1 second. Alternatively boron-ball devices have also been devised. In the case shown a back-up trip signal would occur from fuel thermocouples.

From a transient such as Fig. 4(*b*) a fault temperature rise/fuel rating relationship may be obtained. Then the temperature reached by the hottest can in every channel may be estimated making allowances for random and systematic variations in local power, flow and heat-transfer data. An integrated reactor probability of any channel exceeding, say, the ignition temperature of Magnox or some lower temperature may then be considered as a safety figure of merit.

The long term temperatures would be affected by:

(*a*) Coolant flow and composition (air/CO_2);

(*b*) After-heat (amount and time variation);

(*c*) Graphite properties (some of them partly reactor dependent, such as thermal conductivity, stored energy and oxidation rate); and

(*d*) Graphite operating temperatures.

Air is less effective than CO_2 for core cooling, and can also react exothermally with graphite at working temperatures. The study of the consequences of air passing into the core is therefore important. Estimates

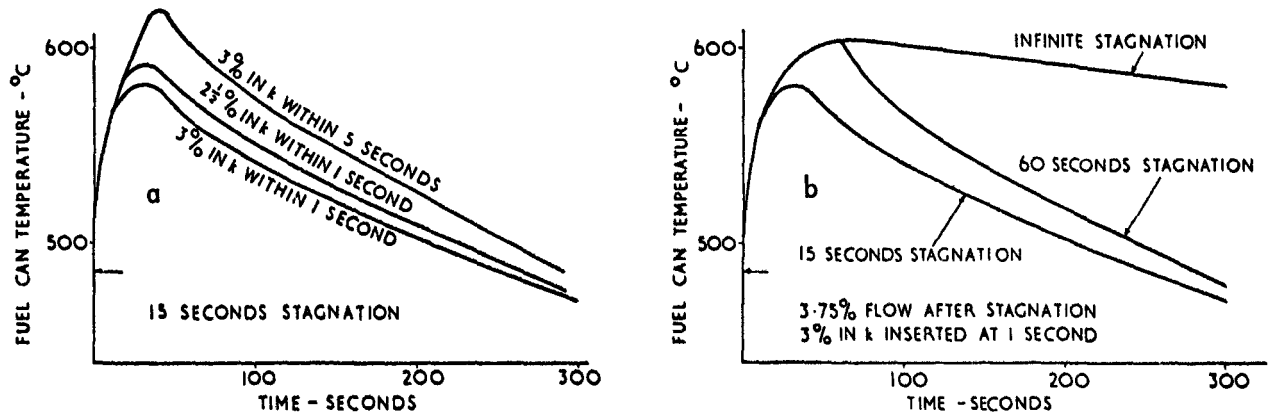


Figure 4. Short term effects of depressurisation
(*a*) Effect of time and amount of absorber insertion.
(*b*) Effect of stagnation time

of ingress for all sizes and locations of duct failure are difficult and depend on the state of the blower and the duct valves in the faulty circuit. Also the extent to which the escaped CO_2 blankets the fracture must have a considerable influence. The essential operator action is to stop the blower and shut at least one valve in the fractured circuit.

All Magnox reactors of UK origin have CO_2 injection equipment to restore a CO_2 atmosphere. A circuit of 6 tons CO_2 capacity at atmospheric pressure requires injection at 20 lb/s to get about 95% concentration of CO_2 in 2 000 s.

For Fig. 5 three out of four blowers have been assumed to be operating with auxiliary drive at 40% speed, which is typical of recent designs. The effect of reaction rate between graphite and air is important and results for three chemical reactivities are shown. Delay of air entry into the core and graphite pores has

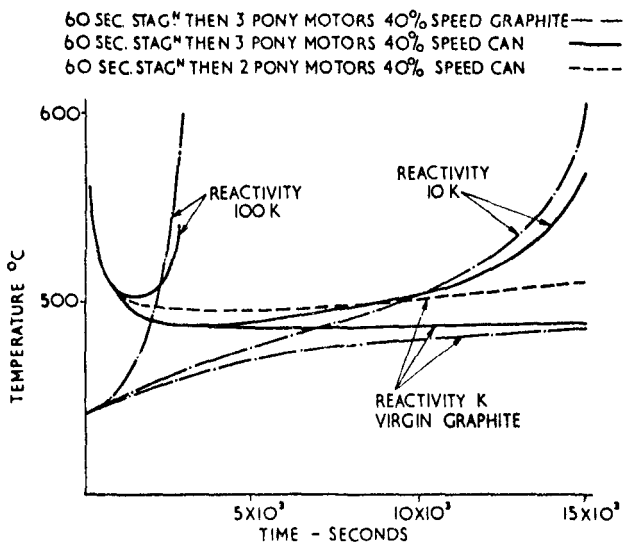


Figure 5. Long term effects of depressurisation (100% air, no CO_2 injection

generally been pessimistically ignored for computation of the curves, as has the influence of diffusional control of the reaction. These problems are discussed in the paper by Beattie *et al.* [9].

Injection of CO_2 can be used to control temperatures for graphite chemical reactivity as high as 100 times virgin value. Research work is proceeding on this property of graphite to discover how it might be controlled, by inhibitors or otherwise.

Collaboration between the Electricity Generating Boards, The Consortia and the UKAEA has been maintained through the reactor design, construction and operations stages to get agreed data and methods of calculation for all safety analysis.

It is however within experience for accidents to happen in the best regulated plants, in unpredicted ways. Accordingly an emergency organisation for plant, site and district action is kept in being at each station. To ensure cooling of the fuel and core the first objectives of plant personnel after an accident involving rapid depressurisation are to:

(*a*) Close at least one valve and shut down the blower in the damaged circuit;

(*b*) Check that the blowers in the other circuits are running at the highest available speed;

(*c*) Restore a CO_2 atmosphere by CO_2 injection if air has entered the core;

(*d*) Operate the fission-product treatment plant in case of need; and

(*e*) Seal the biological shield, whether or not it has been possible to seal the pressure circuit.

Measurements, e.g. of radioactivity due to fission products, could lead to instructions to vary these actions.

NORMAL OPERATIONS

Design and operational controls at commercial nuclear stations limit radiation exposures of personnel

Table 2. Station radiation problems—normal operation

Hazard	Source	Cause	Control (excluding work time restriction)	Remarks
External radiation	Reactor and shield	Fission neutrons and gammas, fission product, capture and activation gammas	Biological shield	Additional control by planned operating routines include manning times and limited access, where necessary. Radiation levels in general, agree with design figures to within a factor 2
	Reactor coolant	^{41}A, ^{16}N and occasionally fission-product gammas	Shielding of ducts (if necessary)	
	Fuel, absorber and control rod handling	Fission neutrons, fission product and other gammas	Shielding of charge M/C	
	Cooling ponds	Fission product and other gammas		
Gaseous effluents [a]	Shield cooling air	^{41}A, ^{16}N and some radioactive Dust	Filtration, monitoring and discharge at high level mixed in shield cooling air	
	Fuelling machine blowdown	CO_2 as toxic hazard, ^{41}A and ^{14}C, fission products, radioactive iron and graphite	Controlled releases above	About one ton daily
	Coolant blowdown	Required infrequently (during maintenance)	Controlled release. Fission products in gas circuit kept low by early removal of failed fuel elements	100–200 tons annually. Measurements during commissioning period required information on CO_2 dispersal from the high level vent
	Coolant leakage	Unavoidable small leakage from pressure circuit	Release mixed in shield coolant air	Of order 1% circuit content per day shield cooling air monitored for CO_2 content, and make-up recorded
Liquid effluents	Cooling pond	Radioactive Magnox corrosion products, fission products	Treatment plant to remove bulk of activity from pond, dilution with condenser cooling water and discharge	Pond water pH maintained to reduce corrosion. Consideration of biological ecological effects of specific isotopes necessary
	Laundry decontamination centre, change rooms etc.	Fission products, radioactive steel, graphite and dust		
	Coolant driers	Tritiated water	Collection and storage in leakproof containers pending disposal	Main source from lithium in graphite-quantity is expected to reduce
Solid wastes [b]	Fuel element components	Radioactive Magnox, fission pro-products	Shielded storage	Highly radioactive materials stored in waste disposal vaults. Some incineration of low level waste where possible
	Control rods, absorber rods	Radioactive steel	Shielded storage	
	Miscellaneous filters, scrap equipment, clothing, swabs, etc.	Fission products, active steel, active dust etc.	Shielded storage	

[a] Subject to Ministry authorisation.
[b] Methods of storage subject to Ministry approval.

in accordance with ICRP recommendations. The main sources of radiation and contamination are summarised in Table 2. As indicated in the table the Licensing Authority (Ministry of Power) or other appropriate Ministries authorise methods of accumulation of radioactive waste materials and the proposed releases. They also keep in touch with the proposed methods for radiological control [10].

Radiological Safety Rules have been applied by the Electricity Generating Boards. These include definitions of radiation and contamination zones, together with specifications for controls at access to such places. Exploratory and routine radiation and contamination measurements are made. Film badges and other dosimetry devices have recorded individual irradiations of staff, among whom are classified workers, subject to regular medical examination. Arrangements permit only the few classified workers to have radiation doses exceeding 1.5 r/year.

Measurements have been made of radioactivity on grass, and in soil and milk samples from near each station, prior to commissioning. These are being continued during operation at the present time. When compared with other measurements made in the UK they have provided assurance that no important deposition of radioactivity is occurring from the station. They have also given the operators experience of making measurements which might be required following an accident.

It has been shown that the hazards from normal operation of gas-cooled reactors for electricity generation are acceptable for both operators and the public.

Commissioning and operating the reactors have yielded useful safety information leading to:

(*a*) Some plant modifications; and

(*b*) Some improvements in technical and administrative procedures.

In the latter category detailed emergency operating instructions for the plant have been developed; also operating rules which have been approved by the Licensing Authority [11]. Personnel training has covered normal and emergency operations. Preventive maintenance schedules and practices for recording data have been developed.

Items of safety significance disclosed during long periods of otherwise satisfactory gas-cooled reactor plant operation are set out in Table 3.

Every effort is made following the discovery of a design or construction weakness to re-establish a standard of safety, modifying as necessary, at all stations. Additional to the day-by-day on-site attention given to safety problems it is intended to reassess comprehensively the safety of the plant as time proceeds.

Further information on the operation of the CEGB reactors is given in reference [12].

DEVELOPMENTS

Magnox and general

During design and construction of the first commercial nuclear power stations in the UK changes have been made by modification, addition and deletion of plant to improve economy and/or safety. The following are typical examples which have been applied to some reactors for the latter purpose:

Item	Safety implication
Supply of automatic sector control rods.	Closer fuel temperature control in normal and fault operation, particularly with regard to stability.
Drop-off control rods.	Depressurisation protection extended.
Fuel thermocouples located near mid-core height.	General improvement in protection.
Fuel thermocouples regrouped radially.	Regional fault protection extended.
Constant fuel temperature trip margin at all temperatures.	Start-up protection extended.
Higher auxiliary blower speeds.	Improved cooling for depressuried reactor.
Guaranteed CO_2 feed.	Control of atmosphere in depressurised reactor.
Biological shield to substantially withstand duct failure.	Improved containment potential.
Fission-product treatment plant.	Earlier treatment of active coolant if necessary.

With higher auxiliary blower speeds, and special attention to the guarantee of essential supplies to drive the blowers, less importance has been attached recently to the early requirements for natural coolant circulation and blower shaft fly wheels.

Integral pressure vessels

Perhaps the most important change of all for gas-cooled reactor safety has been the development of integral pressure vessels. Whereas the layout of the coolant circuits of the first Magnox reactors had boilers outside the biological shield [12] recent designs include the boilers within the pressure vessel. Although what follows refers to a typical concrete vessel design [13] much of it also applies for an integral steel vessel design [14].

The main safety claims, for which realisation is dependent on a particular design, are:

Item	Safety claims
Concrete vessel. (Water-cooled concrete necessarily thick for strength).	Lower external radiation levels. No shield cooling air release. Fission products in coolant less important. Vessel integrity at least as high as steel vessel. Failure mode (if any) gradual and detectable.

Table 3. Operation of UK power reactors—extracts of safety significance

Date	Reactor	Item	Comment	Modifications
Commissioning and early operation measurements leading to plant or operational changes				
1962	BRADWELL [3]	Control rod reactivity worths	Absorber worth greater than design intention	99 rods included in design. Adequate shutdown margin in 64 to get acceptable radial flux distortions. 80 used
1962	BRADWELL [3]	Control rod start up arrangements	Axial flux distortion greater than desirable	Original design single group lift. Two group lift adopted
1962	BRADWELL [3] and BERKELEY	Sector rod worths	Absorber worth and reactivity rate too great compared with design intention	Investment reduced. Gears changed on gear boxes
1962	BRADWELL [3] and BERKELEY	'Local criticality' effect	Withdrawal of 5–7 rods in a particular pattern could lead to criticality	Administrative control for withdrawal in shutdown conditions. Low level flux protection adequate
Plant and other items				
December 1957	CALDER [16]	Disintegration of overspeeded turbo-alternator	Cast iron shot inadvertently left in steam system	Additional steam circuit cleaning on later reactors. Governor and runaway stop valve guides hardened. Protection for vital plant items reviewed
May 1959	CALDER [16]	Voltage disturbance stopped blowers and delayed operation of reactor trip circuits. Short 20 °C fuel can transient experienced	Caused by build-up and probably intermittent breakdown of a high resistance film on contact surfaces of the carbon brushes of an alternator pilot exciter	Routine examination of alternator brushgear. Protection system for blower motors modified
September 1961	CALDER [16]	External grid voltage disturbances during thunderstorm caused blowers to slow and trip. Reactor tripped	Caused by slow opening of grid circuit breakers	Circuit breaker operating speed increased. Thunderstorms warnings passed to station control
January 1963	AGR [4]	Collapse of hot duct stainless steel insulation liner. Circulator surged. Reactor tripped	Inadequate strength of liner tube	Local repairs
November 1963	AGR [17]	Irradiated fuel stringer misplaced in unshielded test area. High gamma level. Reactor shut down		Pending (January 1964)
October 1962	BERKELEY	Burst cartridge detection gear box fault		Redesign. Periodical replacement of gearing
March 1963	BERKELEY	Blower master selsyn relay fault	Blower speed increase caused reactivity addition (negative fuel element temperature coefficient)	Component fault corrected
January 1963	BRADWELL	Sporious tripping successively of both reactors due to grid voltage surge in extreme weather conditions	Operator action correctly controlled plant performance	Modification to plant operating procedures

	Missile and fire proof.
No main gas ducts.	Duct failure eliminated.
Few large penetrations exceeding 18 in diameter.	Small (if any) conceivable penetration failures.
	Depressurisation fault transients much reduced.
	Air entry insignificant subsequent to depressurisation.

The absolute level of safety attained with any system cannot be quantitatively defined and any supposed increase claimed for integral concrete or steel vessels need not necessarily be significant in relation to the high degree of safety inherent in well designed non-integral steel vessel systems. Relaxation of siting requirements for the latter systems might well have come with the passage of time and the accumulation of operating experience. The integral pressure vessel has given an earlier opportunity for deliberate revision of siting intentions based on claimed engineering improvements. If the freedom from rapid depressurisation can be established the possibility arises of an increase in fuel rating.

Advanced gas-cooled reactors

The commercial advanced gas-cooled reactors following Windscale AGR will benefit from the design work done on earlier Magnox reactors.

The control and protection system required is similar to Magnox reactor practice. The most significant changes are the higher coolant outlet temperature and the higher fuel rating, values for the latter of up to 20 MW/t being possible. Also important is the use of enriched uranium oxide in stainless steel in place of natural uranium metal in Magnox for fuel.

Stainless steel does not react rapidly with either air or CO_2 at any temperature up to its 1 460 °C melting point. In that range of temperatures UO_2 oxidises slowly in air, but not in CO_2. At higher temperatures it is stable in air and CO_2 [15]. There is thus no danger of a fuel fire occurring in or being propagated along an AGR channel.

The long-term corrosion effects of CO_2 on stainless steel may set a can surface temperature limit in the range 800–900 °C. The nominal operating temperature is likely to be nearer 700 °C.

To control the progressive release of volatile fission products into the can-fuel interspace a centre UO_2 temperature limit in the range 1 600–1 800 °C may be utilised. The gas pressure inside the cans should then be sufficiently below coolant pressure, at the appropriate temperature, for out-leakage of fission products to be slight, even with a pin-hole can defect. As UO_2 will not swell in CO_2, pin-holed cans should remain relatively undamaged during irradiation even if in-leakage occurs. The principle of a fission-product retaining fuel, and consequently very low levels of fission-product activity in the coolant, is thus retained.

Systematic failure of cans may be expected if high can temperatures (over 1 000 °C) persist long enough for creep of the steel to occur due to excess differential pressure across the can wall. Burst cartridge detection monitoring will detect the onset of any such can deterioration before more than a few cans are affected.

AGR fuel has about the same negative temperature coefficient as Magnox fuel. For fuel at maximum irradiation the moderator temperature coefficient is expected to be only slightly positive. The greater permissible rise between operating and failure temperatures eases the problem of detecting and correcting faults. Also there is improved stability against reactivity accidents and more scope for faster start-up if this is required.

Although the power density of an AGR is higher than for Magnox reactors it is still low when compared with many power reactors. The graphite is still a very effective heat sink. In a typical transient where all blowers are assumed to run down to pony motor speed the nominal maximum fuel can temperature is unlikely to exceed 900 °C. In the short duration of such a transient fuel can failure is unlikely. No failure mechanism is known for fuel maintained at this temperature for up to an hour.

Release of volatile fission products from cans during coolant depressurisation may be avoided by arranging the balance between heat production and heat removal so that the fuel can walls are always sufficiently strong to withstand any excess pressure from within.

The high graphite sleeve temperature makes the prevention of air ingress more important than on Magnox reactors. However, significant ingress can probably be prevented with an integral vessel design for any credible mode of depressurisation, possibly with the assistance of CO_2 injection.

CONCLUSIONS

In retrospect, by study of hypothetical accidents for the UK Magnox reactors, the decision taken early in the design of Calder Hall to omit any special containment has been confirmed as a sound one.

The biological shields have been developed so that, together with other equipment provided to shut down the reactors and restrict the effect of air ingress, a wide range of loss of coolant accidents can be controlled.

The low probability of some accidents studied for these reactors would have ruled them out from further consideration in most conventional industries; however, improvements to plant and operating procedures have in many cases been applied rather than accept these small risks. Operations at the power stations have been fairly trouble free in the nuclear plant and

they have not disclosed reasons for always being so conservative.

It is possible that detailed study of specific, more highly rated Magnox and AGR designs of the future will show that these also require no special containment. The advent of the integral pressure vessel makes this more likely and siting relaxation for designs using this feature may in due course be claimed.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/183 Royaume-Uni

Sûreté des réacteurs refroidis par un gaz au Royaume-Uni

par G. R. Bainbridge *et al.*

Les progrès effectués au Royaume-Uni au cours de la période 1954-1964 dans la mise au point du réacteur modéré au graphite à réfrigérant gazeux ont été tels que ce réacteur en est arrivé au stage où la production d'électricité est devenue pratique, avec des normes économiques et de sûreté comparables à celles des installations non nucléaires.

Le mémoire décrit les progrès dans la conception des centrales nucléaires en Grande-Bretagne et leurs effets sur leur sûreté. Au début de la décennie en cause, seules les piles à refroidissement à l'air à circuit ouvert de Harwell et de Windscale étaient en service. Mais à la date de la Conférence de Genève de 1958, plusieurs réacteurs refroidis au CO_2 en circuit fermé étaient en fonctionnement à Calder Hall, avec un combustible d'uranium métallique gainé de Magnox.

L'expérience très utile concernant le fonctionnement et la construction de ces réacteurs, ainsi que celle concernant les centrales nucléaires civiles, a conduit à la mise au point rationnelle du réacteur perfectionné à réfrigérant gazeux (AGR), avec combustible d'oxyde d'uranium gainé d'acier inoxydable. Le réacteur expérimental perfectionné à réfrigérant gazeux de Windscale fonctionne à pleine puissance pour la production d'électricité depuis plus d'une année et la première grosse centrale de ce type est à l'étude. Grâce à l'expérience de fonctionnement et de construction on pourrait être plus libéral dans la réglementation concernant l'implantation de nouvelles centrales au Royaume-Uni.

Pour chaque centrale nucléaire existante, on a effectué des études intensives concernant la sûreté, y compris des analyses des défaillances possibles du genre décrit dans le mémoire. Une collaboration étroite a été maintenue entre les consortiums, les directions de l'électricité, les pouvoirs publics et l'UKAEA pour analyser et comprendre les caractéristiques générales et spéciales de sûreté de chaque centrale et pour résumer les résultats de cette analyse dans les rapports sur la sûreté de chaque centrale. Cette étude des problèmes de sûreté, depuis le moment du choix initial de l'emplacement jusqu'à celui du fonctionnement à pleine puissance, continuera pour chaque nouvelle centrale à une échelle correspondant à ses caractéristiques particulières.

De nombreuses améliorations sont intervenues dans la conception, dans la compréhension de l'équipement nécessaire en cas d'urgence, et dans les méthodes d'exploitation; cela est un résultat direct des analyses de sûreté et le mémoire en donne des exemples types.

A/183 Соединенное Королевство

Обзор работ по безопасности английских реакторов с газовым охлаждением

Г. Р. Бейнбридж *et al.*

В период с 1954 по 1964 год в Великобритании велась успешная разработка реакторов с газовым теплоносителем и графитовым замедлителем. В результате этой работы стало возможным получать электроэнергию для практических целей, при этом экономика и безопасность атомных электростанций сравнимы с неядерными системами.

В докладе описывается прогресс в разработке конструкции ядерных электростанций Великобритании и его влияние на безопасность этих станций. В начале указанного десятилетия эксплуатировались два прямоточных реактора с газовым охлаждением в Харуэлле и Виндскейле. В 1958 году, в период работы женевской конференции, эксплуатировалось уже несколько реакторов с замкнутым циклом охлаждения углекислым газом в Колдер-Холле. В качестве топлива этих реакторов используется металлический уран с магноксовой оболочкой.

Результаты проектно-конструкторских разработок, опыт эксплуатации и практика сооружения действующих ядерных электростанций гражданского назначения были использованы при разработке усовершенствованного реактора с газовым охлаждением с топливом из двуокиси урана в оболочке из нержавеющей стали. Виндскейлский экспериментальный усовершенствованный реактор с газовым охлаждением проработал более года на мощности и показал возможность создания первой крупной ядерной электростанции такого типа, которая в настоящее время разрабатывается. Опыт разработки и эксплуатации станций показал возможность смягчения требований, предъявляемых к вновь разрабатываемым ядерным электростанциям Великобритании.

Всесторонне исследовались вопросы безопасности каждой станции, включая анализ аварийных ситуаций, подобных описанным в докладе. Имело место тесное сотрудничество между Консорциумом фирм, энергетическими управлениями, Лицензионным управлением и Управлением по атомной энергии Великобритании в проведении анализа общих и специальных характеристик безопасности каждой станции и в обобщении результатов этого анализа в докладе о безопасности работы отдельных станций. Такой анализ проблем безопасности будет производиться для каждой новой станции с момента выбора строительной площадки до пуска станции на полную мощность.

Существенно усовершенствованы конструкция станций, методы эксплуатации, и разработано оборудование, которое необходимо для аварийных случаев. В качестве примера в докладе приводится расчет типичных случаев.

A/183 Reino Unido

Examen de la seguridad de los reactores refrigerados por gas del Reino Unido

por G. R. Bainbridge *et al.*

Durante el período 1954-1964, el desarrollo de los reactores refrigerados por gas y moderados por grafito ha avanzado en el Reino Unido hasta una fase en la que es ya práctica la producción de electricidad, con niveles económicos y de seguridad comparables con otros sistemas no nucleares.

Esta memoria expone el desarrollo progresivo del proyecto de centrales nucleares británicas y su efecto en la seguridad de las mismas. Al comienzo del último decenio solamente estaban en funcionamiento los reactores de Harwell y Windscale, refrigerados por aire en circuito abierto, pero en la época de la Conferencia de Ginebra de 1958, ya estaban en explotación en Calder Hall varios reactores refrigerados por CO_2 en circuito cerrado, que empleaban como combustible uranio metálico con vaina de Magnox.

La valiosa experiencia adquirida en el proyecto y explotación de aquéllos y de las muchas centrales nucleares comerciales construidas desde entonces, han cristalizado en su evolución lógica que es el advanced gas-cooled reactor (AGR), que emplea como combustible óxido de uranio con vaina de acero inoxidable. El AGR experimental en Windscale ha funcionado a plena potencia durante más de un año y se está proyectando la primera gran central comercial de este tipo. Dada la gran experiencia acumulada en el proyecto y explotación de centrales nucleares, podría permitirse actualmente a las nuevas que se instalen en el Reino Unido una relajación importante en los criterios de emplazamiento.

Para cada central se han realizado amplios estudios de seguridad, incluyendo análisis de fallos de los tipos señalados en esta memoria. Ha habido una estrecha cooperación entre las diferentes partes interesadas — consorcios, compañías eléctricas, autoridades competentes y la UKAEA — para analizar y comprender las características de seguridad generales y específicas de cada central y para resumir los resultados de este análisis en los informes de seguridad individuales. Este estudio de los problemas de seguridad, desde la fase de elección previa del emplazamiento hasta la explotación a plena potencia, continuará

para cada nueva central en la escala que pidan las condiciones particulares de cada una.

Como consecuencia de los estudios de seguridad se han conseguido considerables mejoras en los proyectos, en el conocimiento de las necesidades de equipo para situaciones de emergencia y en los métodos de explotación, presentándose en esta memoria una descripción de ejemplos típicos.

US reactor experience and power reactor siting

By C. K. Beck *

More than 21 years have passed since the first nuclear reactor achieved criticality. Several hundred reactors, of a wide assortment of size and types in many countries all around the world, have been built. The expanded understanding of reactor characteristics and behavior and the continuing accumulation of successful operating experience have led to questions about the degree of conservatism or possible over-conservatism in our present presumptions and practices in the safety design and location of reactor facilities.

From the outset of the atomic energy endeavor, there has never been any substantial disagreement that nuclear power reactors in general present very large inherent hazards to the general public. The spectacular nuclear explosions that marked the advent of the nuclear age left an indelible impression of dangerous capability associated with anything nuclear, including reactors, in the minds of reactor designers and of the public alike. These unusual circumstances accompanying the birth of this new technology, as well as certain unique and unfamiliar characteristics of radiation hazards, have resulted in preoccupation on the part of both laymen and expert with the potential possibilities of damage from nuclear reactors (often indiscriminately classed with atomic bombs) to an extent unprecedented in any other technical field.

In contrast, in other areas of technology the concern for safety has commonly been with the practical exhibition of hazards, or lack of safety. That is, experience and the actual level of damage demonstrated in practice, not the theoretical potential for possible or conceivable possibility of damage, has set the level of standards and requirements and concern for safety.

Nuclear reactors, having been "born" under the constraints of safety requirements dictated by considerations of the theoretical potential for hazard, are just now or shortly will be reaching the point where some reappraisal of the "base line" of hazard, or the margins of safety under present practice may be possible. In some instances in other areas of technology, imposition of obviously needed safety requirements was delayed until serious accumulations of actual damage forced remedial measures, whereas in the absence of traditional practice, earlier actions may have been taken. For nuclear technology, the inertia of present practice may work in reverse. It is the most natural course to continue present practices, particularly when these have resulted in an extremely high level of safety. More importantly, before any established safety patterns can be altered, a basis must be laid for showing that revised practice will not diminish what has been thought to be a necessary degree of protection to an unacceptable level. On the other hand, pressures are at hand for re-appraisal and certain factors have evolved which contribute toward the feasibility of such re-examination. Among these:

(*a*) The proximity of economically competitive nuclear reactor applications give a healthy incentive to critical examination of every component and requirement.

(*b*) The continuing and expanding safety research and engineering test programs of the Atomic Energy Commission are expected to resolve shortly some of the uncertainties in reactor technology that force assumptions to be made without a firmly established technical base. As knowledge is gained on such items as transient reactor behavior, fission product characteristics, life expectancy and dependability of structural components, etc., the analyses and evaluation of reactor hazards become more realistic and hence, the necessary measures of protection may become more exactly defined. Likewise, the reliance that can be placed on protective safeguards become better established. (The description of US nuclear safety research is listed elsewhere on the agenda of this conference.)

(*c*) Publication, two years ago, of the United States Guides to Selection of Sites for Power Reactors established for the first time a coherent indication of the essential factors that must be considered in selecting power reactor sites. These factors then became targets for critical analysis and evaluation relative to distance requirements in reactor sites. An impetus was thereby given to the re-examination of the whole matter of assumptions, presuppositions, and relationships between the complex technical factors of importance.

The status of site selection guides and practice in the United States is discussed in greater detail below.

The accumulation of substantial operating experience has shown that damage resulting from operation of nuclear reactors up to the present time falls far

* US Atomic Energy Commission, Washington, D.C.

below the theoretical upper limits. At some point, sufficient experience will have been accumulated from the record of reactors operated, and accidents experienced, to establish an actuarial basis for estimating the practical magnitude and consequences of reactor hazards. We have not, in my opinion, fully reached that point as yet, though the actual accumulation of operating experience is more extensive than might be generally realized.

The records of operating experience and the significant accidents that have occurred are discussed later.

THE STATUS OF SITE SELECTION CRITERIA

A. Presuppositions

The possibility that radioactivity accumulated in nuclear reactors might be released and find its way in dangerous amounts to populated areas is the only significant threat to the health and safety of the public which arises from these facilities. Such releases might occur by normal procedures in routine operations, or they might occur by accident.

It is the latter possibility that gives rise to the difficult decisions that must be made in selecting sites for nuclear power reactors. Radioactive effluents are routinely released into the environment from most power reactors and the intended practice for any given reactor does affect the choice of site for that facility. However, there is a wide range of optional choices available to the owner, which basically relates to questions of economics, not safety. By design and procedures, and at a cost, routine radioactivity effluents can be limited to whatever extent might be required at a particular site; might in fact be virtually eliminated. The reactor owner has the latitude of a wide choice in balancing the extensiveness of control against the characteristics of site.

The matter is altogether different for the possibility of accidents. Unintended, inadvertent, accidental releases, if any should occur, are by definition unpredictable. Neither their magnitude nor time of occurrence can be foretold. This inherent fact is the heart of the problem. In every power reactor the accumulated inventory of radioactivity is sufficient to result in widespread sickness and death in surrounding areas if it were released to the environment in an uncontrolled manner.

There are only two basic premises which would permit operation of facilities which possess hazards of such dimensions: (*a*) Isolation distance: The facility could be located sufficiently far from inhabited areas that serious danger could be averted by dilution and dispersion of the escaped materials before they should reach inhabited areas; or (*b*) Containment: Design features and safety devices which would prevent the escape of dangerous amounts of radioactivity could be provided. (In past practice, dependence has been placed on a combination of these: some isolation distance and some engineered safeguards.)

Engineered safeguards, i.e., those components or systems intended specifically to prevent the accidental release of radioactivity inventories are of two types: (*a*) those which would prevent serious accidents (i.e., fission products-releasing accidents), and (*b*) those which would prevent escape to the environment in case a serious accident should occur.

In principle, it should, and in time it may, be possible to place sufficient confidence in the knowledge of reactor behavior, the design of the reactor and the dependability of "accident-prevention" safeguards to believe that a serious accident could not or would not occur. It may develop with further operating experience and the results of extensive observations in hand of what happens when reactors are badly damaged, either in our planned experimental program or otherwise, that we will conclude with confidence that failure, malfunction and misoperation will not lead to serious accidents. But dependability on this expectation alone can hardly be justified at the present stage of reactor development and experience when the reactor is located in the midst of people and the consequences of a misjudgment could be so severe.

Thus, in the meantime, and in all cases up to the present time, some combination of isolation distance and "consequences-limiting" safeguards have been provided.

B. Essential principles of the site guides

From basic considerations such as the above, and from observation of the general practice of reactor siting as it had been practised up to that time,* a set of "Reactor Site Selection Guides" was developed and was published in 1961.

The basic elements in these guides included the following:

(*a*) The guides identified the important factors to be considered in selection of reactor sites.

(*b*) The guides provided the framework for analysis and evaluation of any particular site though recognizing that quantitative and unique determinations of "safe" and "unsafe" distances cannot be made and a certain degree of arbitrariness or subjective judgment is finally inescapable in some of the elements involved.

* Before publication of the site guides, the decision on suitability of sites for proposed reactors was made by way of detailed study of each proposed project on a case-by-case basis by the Commission's technical staff and by the technically expert members of the Advisory Committee on Reactor Safeguards. Many considerations included in the earlier case studies were incorporated as part of the coherent and systematized procedures articulated in the site guides. The guides were published to inform prospective reactor builders of the factors which would be considered in deciding on their proposed facility and its location. Decisions are still made on case-by-case studies.

(*c*) The concept of a maximum credible accident, for each reactor facility, i.e., the upper limit of hazard (fission product release) against which the protective characteristics of the site and the safeguards of the plant must be compared, is taken as the point of departure for the evaluation of site suitability and plant design.

(The concept of the maximum credible accident, i.e., the upper limit of practical hazards, evolves from the following: if it can be assumed that no radioactivity-releasing accident will occur, then the question of site selection is virtually academic; almost *any* site is acceptable, safety-wise. On the other hand, if it must be assumed that the maximum conceivable accident can happen, i.e., release to the environment of all the accumulated radioactivity, then virtually *no* sites are far enough away from inhabited areas to be acceptable. There is left then the question: "What is the maximum credible accident, the upper limit of practical hazard, considered sufficiently likely to occur that it must be taken into account in considering site characteristics?")

(*d*) The site guides establish the concept of, and specify numerical values for, potential radiation exposure doses to be employed as measuring indices in evaluating reactor site characteristics, i.e., 25 rem whole body exposure or 300 rem to the thyroid. These are not permissible emergency doses; they have no relationship to protective actions to be taken after an accident. Instead, they are comparative measuring indices for use in evaluating the adequacy of reactor safeguards and site characteristics.

(*e*) The guides make explicit, and define, the concepts of population zones around a reactor which had been observed in practice all along; the exclusion zone, where people, if any, are highly mobile and under the direction of the reactor operator in emergencies; the next succeeding low population zone, where evasive or protective measures for people could be taken in case of dangerous releases from the plant; and the city distance, where people are not very mobile, but the distance is sufficient to prevent the worst aspects of catastrophe should a serious accident occur.

(*f*) The guides, in defining how the magnitudes of zones may be determined, establish the principle that safety, in case of accident, depends not only on distances but also on protective features of the facility — the "engineered safeguards" of the facility: "where unfavorable physical characteristics of the site exist, the proposed site may nevertheless be found to be acceptable if the design of the facility includes appropriate and adequate compensating engineering safeguards".

C. "Engineered safeguards"

Every nuclear reactor has particular and often additional features as required by safety incorporated in many of its basic components. It is often impossible to distinguish between the safety features and safeguard devices included in the basic design of a plant and those additional and unique items which would fall within the special safeguards, particularly for the "accident prevention" category.

Nevertheless (perhaps arbitrarily), certain items are usually considered when "accident prevention" safeguards are mentioned:

(*a*) Back-up or emergency reactor shut-down systems;

(*b*) Poison "fuses";

(*c*) Emergency cooling systems;

(*d*) Emergency electric power supplies;

(*e*) "Last-ditch" injection or other systems to prevent melt-down of the core after loss of coolant.

Similarly, "consequences-limiting" safeguards include:

(*a*) Containment (the low leakage "barrier of last resort");

(*b*) Spray, wash-down system to quench steam or reduce hot gas pressures and wash down soluble fission products;

(*c*) Internal filtration system to collect fission products;

(*d*) Double containment with "pump back".

When such systems are included in the design of the reactor, there are three basic questions to be resolved relative to their effectiveness in each case:

(*a*) What effect will the safeguard system have if it operates as intended?

(*b*) What would be the consequences of its failure to operate?

(*c*) How dependable is its performance likely to be?

The first two of these questions in general can only be answered by analysis of the situation in each particular reactor facility. The answer to the third question, the dependability of operation, must be "very high indeed" when a safeguard is to be used in lieu of distance. Certain criteria have been listed which should be included when the level of dependability of a safeguard is being considered.

(*a*) Performance specifications must be established, and must be met;

(*b*) Performance specifications must be met with the equipment in the final installed condition.

(*c*) The performance specifications must be met at all times;

(*d*) The performance specifications must be met under accident conditions;

(*e*) The safeguards system must apply to the whole system (e.g., if part of the primary coolant system

lies outside of the containment barriers, the assumed protection may be compromised unless there are other special barriers in the potential escape routes).

(*f*) There must be constant vigilance against deterioration with age of little-used safeguards equipment.

(*g*) The capability of systems should be established by test.

(*h*) Test should be carried out under conditions relevant to accident situations.

(*i*) Safeguards systems should be periodically re-tested where possible.

(*j*) The safeguards systems must be protected from damage by the accidents which give rise to the demand for their services.

(*k*) If the systems work as intended, there are often unexpected "follow through" problems. (For example, in one case, where a high efficiency filter system had been designed and the designer had calculated how much fission products it would collect, the fact had been overlooked that decay heat from so much radioactivity would not only have melted the filters but would have vaporized them!)

D. Seismic problems in reactor site selection

In the site guides the problem of earthquake effects on reactors in seismically active areas is flagged as an item to be considered by the statement: "... No facility should be located closer than one-fourth mile from the surface location of a known active earthquake fault..."

In an actual site selection case in a seismic area, the problem is considerably more complicated than this statement indicated. The state of our knowledge about earthquakes, their probabilities of occurrence, the vibration patterns and forces they produce, how these vibrational forces are transmitted and attenuated, and the probability of surface ruptures, is sadly deficient with respect to quantitative measures of seismic effects on reactor site selection. In consequence, considerable uncertainty exists in the application of these factors to reactor location and design.

Earthquake phenomena

Tectonic earthquakes are thought to result from slow, localized stress build-up within proximate layers or areas of the internal earth's crust in consequence of the general non-static, dynamic character of the earth's crust itself. Stress builds up in the interface area between the layer which is moving and the stationary one, but slippage may not occur because of friction and shear strength of the materials. When the stress, which may extend along a lengthy internal layer, at some point exceeds the stress resistance, slippage occurs, and the movement may be propagated over the whole stressed surface. Such movement usually consists of an exceedingly complex assortment of starts, jerks and stops at localized places over the whole surface movement. When the whole erratic, jangled, disharmonic movement has been completed, a net lateral (vertical or, more often, horizontal) displacement may have taken place, amounting in some cases to a displacement of many feet at the earth's surface (up to 21 feet displacement in the 1906 quake in California).

Earthquake effects relative to reactor safety

Two consequences of earthquakes are of significance to safety of any nearby reactors: (*a*) Vibrations and shaking of the ground, and (*b*) Faults, i.e., differential movement and displacement of adjacent areas of the ground.

It is generally agreed that, by suitable design, structures can be constructed to withstand the vibration effects of earthquakes. From information now available, it does not appear feasible to design structures to withstand significant faulting more than a few inches to a foot or so. Areas where there is a large likelihood of extensive faulting are not suitable for reactor sites.

To design against vibrational effects, the "dynamic approach" must be used. Four steps are necessary: (*a*) The vibrational pattern to be anticipated in the maximum future earthquakes must be established, including the expected periods, amplitudes and accelerations. (*b*) The natural vibrational characteristics of the structure or component in question and its damping factors must be established. (*c*) The limit of "working stress", acceptable in the material of structure or component, due to normal loads plus the stress from earthquakes must be selected (e.g., one-half or one-third of yield). (*d*) The structure must be designed so that the working stress will not be exceeded if an earthquake of the magnitude and vibrational spectrum of the one assumed should occur.

To estimate the location and characteristics of future earthquakes for purposes of reactor design, there appears to be no better basis than to assume that the seismic pattern in an area over the recent past (a few hundred years) will likely continue in the immediate future. On this basis, assumptions will be that future earthquakes will have magnitudes as large and vibrational patterns the same as the largest known in the region in the past; surface ruptures will tend to occur along old fault lines; faults will appear where they have occurred in the recent past.

E. Present status and trends in reactor siting

In the report on "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear

Table 1. Commercial and public power reactors

Name	Type	Power MW(th)	Exclusion distance (miles)	City distance (miles)	Site approved
PWR	Pressurized water	231	0.4	7.5	before 1955
Con. Edison.	Pressurized water	585	0.3	17.0	5/55
Dresden	Boiling water	630	0.5	14.0	7/55
PRDC	Sodium graphite	300	0.75	7.5	7/55
Yankee	Pressurized water	485	0.5	21.0	7/57
Elk River	Boiling water	58	0.23	20.0	8/58
Piqua	Organic	48	0.14	27.0	1/59
Carolinas-Va.	D_2O; Tube type	63	0.50	25.0	1/59
Hallam	Sodium graphite	240	0.25	17.0	7/59
Saxton	Pressurized water	20	0.17	20.0	9/59
Pathfinder	Boiling water	203	0.5	3.5	12/59
Big Rock	Boiling water	240	0.5	135	3/60
PG & E (Humboldt)	Boiling water	202	0.25	3.5	3/60
Bonus	Boiling water	50	0.25	12.0	3/60
Peachbottom	Gas-cooled	115	0.57	21.0	3/60
Malibu	Boiling water	1 473	0.11	13.0	6/60[a]
S. Cal. Edison	Pressurized water	1 150	0.19	17.0	7/60
La Crosse	Boiling water	165	0.21	19.0	12/62
Conn. Yankee	Pressurized water	1 473	0.25	9.5	2/63
PG & E (Bodega)	Boiling water	1 008	0.50	21.0	4/63[a]

[a] Site tentatively approved except for the currently unresolved earthquake question.

Power Plants" * written in early 1957, before any large nuclear electric power plants began operating, it was assumed, for purposes of calculation, that such reactors would be 30 miles from a large city, because "all the power reactor sites proposed to date are within 30 to 40 miles of a city...". Since that time, 12 large licensed power reactors have been built and sites have been approved for 8 additional licensed reactors (Table 1). These reactors range in power level from 20 to 1 475 MW(th). They are located on sites of from 3 to 2 000 acres, and at distances of 3.5 to 135 miles from population centers of 20 000 or more people (some of the site dimensions and distances from cities were chosen by the owners for reasons other than safety regulation requirements). Approval of these sites was spread over the period of more than 8 years preceding 1964.

For large power reactors, the practice has been followed, in all cases, of requiring a minimum distance of several miles from a center of population. Since publication of the site guides, sites for three large reactors (Southern California Edison, City of Los Angeles ** and Connecticut Yankee, the three largest private power reactors in progress) have been approved with less distance requirements than would otherwise have been required had not certain specific safeguard systems been incorporated into their design.

It is likely that some reduction of distance requirements in site location of future power reactors may continue. How rapidly this trend will continue depends on the confidence that can be developed in the dependability of safety systems, which have in general yet to be developed to their full potential for reliability, and on the experimental and operational record of performance of reactors and reactor safeguards.

OPERATIONAL EXPERIENCE WITH REACTORS

A. Operating statistics

A total of about 246 reactors have been built and operated in the United States (Table 2) *, not including critical facilities.

Table 2. Approximate reactor operating data

Category	Number	Reactor-years [b]
Commercial and military power reactors	16	29
Propulsion reactors	52	127
Production reactors	14	155
Experimental, test, research, teaching [a]	164	718
Total	246	1 029
Critical facilities [a] 35	71 (cells)	292
Total	317	1 321

[a] Certain small temporary facilities in the early days may have been overlooked.

[b] Years in operating status.

* The Possibility and Consequences of Major Accidents in Large Nuclear Power Plants, WASH 740, USAEC, March 1959, Supt. Documents, US Government Printing Office, Washington, D.C.

** Site tentatively approved except for a question of landslides currently unresolved.

* Certain small temporary facilities in the early days may have been overlooked.

Most operating experience has been accumulated with water cooled and moderated reactors, though many other types are included: graphite moderated, gas-cooled, sodium cooled, heavy water, fast, organic moderated and cooled; uranium fueled, plutonium fueled, thorium-uranium fueled, etc. Reactors have been operated for a wide variety of purposes; isotope-production, teaching, research, experimental, test, power-generation, vehicle propulsion, medical irradiation, and space heating. Assorted organizations, with widely differing operational procedures and philosophies, have been responsible for the operation of these 246 reactors: governmental laboratories, industrial companies, military departments, universities, private research organizations, hospitals.

These 246 reactors of widely varying types and characteristics have been operated under a broad range of circumstances and conditions for a total accumulated time of over 1 029 reactor years (Table 2) up to the beginning of 1964. If the critical facilities are added, the total facilities number 317, and the accumulated operating time is 1 321 reactor years. Continued operation by presently operating reactors is currently accumulating 0.6 reactor-years, or 11.5 MW-years of operating experience, each day.

This record of operation has not been accumulated without a variety of unplanned incidents and some serious accidents. But the significant accidents have been remarkably few and only one has had consequences which exceeded more than a small fraction of the theoretically potential or conceivably possible hazard from the reactor involved.

B. Reactor accidents and incidents

Failures of components or equipment; malfunction of instruments or other devices, errors of operators, deficiencies in design, these inevitably and unavoidably occur in any facility as complex as a reactor system. Events or instances in such categories, if not of daily occurrence, are sufficiently frequent to be a familiar experience. Most of these are trivial, and of no consequence, or correction is made without harm. For example, there is an extensive record of material failures of many kinds. None of these caused accidents, but some would fall into the category of "near misses". As examples:

(*a*) In one reactor, 2 adjacent stud bolts on the reactor head were cracked from stress corrosion;

(*b*) In another reactor, 3 (non-adjacent) stud bolts were cracked;

(*c*) In one large power reactor, 2 main control rod shafts were badly cracked from stress corrosion;

(*d*) In three separate reactors, cracks more than half way around on the primary steam or water coolant line were discovered.

Occasionally, however, some failure, malfunction, misoperation, or design deficiency alone, or in combination with others, lead to serious consequences.

In this discussion, a reactor accident (or incident, the less serious events) is taken to be any occurrence in an intentionally sustained chain-reacting, steadily operating system which results in damages to the system, over-exposure or injury to people, or significantly abnormal power behavior. Mishaps in critical facilities, and events of inadvertent criticality in fuel processing systems are excluded. Simple failures of fuel element cladding and fission product leakage or other components are also excluded as being in the normal expectancy category, unless these lead in turn to more extensive consequences. Reactors deliberately intended for destruction are excluded, except for two cases where substantially unexpected results were achieved.

Twenty accidents and incidents,* which fall within the definition above, have been tabulated for the 246 reactors over the 21 year period since initial criticality in the first reactor. These have mostly been described in published reports.

Categories relating to the principal initiating causes are listed below. In many cases secondary or contributory causes, e.g., improper operator response to the first signals of trouble, may have had substantial influence on the course of the accident.

(*a*) Blocked coolant flow channels: 3 instances; some fuel damage in each case. No injury.

(*b*) Pocket of accumulated hydrogen, explosion, fire: 1 instance: damage to auxiliaries. No injury.

(*c*) Improper manual manipulation of control rods: 2 instances; in one, the operator received 2.5 r over-exposure; in the other (SL-1), 3 operators were killed; the reactor was very badly damaged.

(*d*) Instrument failure or inadequacy: 3 instances; 250% over-power in one, but no damage; small melt-down of fuel in the second; extensive fuel melt-down in the third. No injuries.

(*e*) Continued operation under poor conditions; impurities in coolant: 1 instance; extensive melt-down of fuel. No injuries.

(*f*) Interaction of experiments with reactor behavior: 2 instances; overpower and erratic power level. No damage, no injuries.

(*g*) Exploratory operation in known hazards area of reactor behavior: 2 instances; in one, where defective fuel was contributory, small fuel melt-down and some gaseous radioactivity release; in the other, extensive fuel melt-down. No injuries; no personnel exposure; no area contamination.

* A few minor accidents in experimental reactors in the early days, when records were not as extensive as now, may have been overlooked.

(*h*) Over-exposure during refueling with inadequate shielding: 1 instance; 8 persons over-exposed from 2 to 20 r.

(*i*) New operator observing wrong instrument during start-up: 1 instance; over-power, some fuel damage. No injury.

(*j*) Erroneous calculation of rod start-up patterns: 2 instances; over-power and some fuel damage in one; too rapid power rise in the other. No damage; no injury.

(*k*) Miscalculation of effects in deliberately destructive experiment: 2 instances; extensive damage beyond that intended. No injury.

From examination of these accidents in the context of the logged operating time of 246 reactors, the following observations are pertinent:

(*a*) Accidents can happen.

(*b*) Accidents happen from unanticipated causes.

(*c*) Accidents often cause unexpected results.

(*d*) Accidents occur at unexpected times, but usually under abnormal or unusual conditions.

(*e*) Accidents may inactivate the safeguards which are depended on for protection in case of accident.

(*f*) The large majority of accidents (16 out of 20) have occurred in experimental, test and research reactors; only 4, all minor, in power, propulsion and production reactors.

(*g*) Consequences of accidents thus far have been in general much less severe than would have been expected from the accident conditions and circumstances.

(*h*) A "maximum credible accident" may occur, though nothing approaching an accident of this magnitude has thus far occurred, except possibly for the small SL-1 reactor, where damage occurred approaching what may have been estimated as the maximum credible accident.

(*i*) Over-all accident experience is extremely favorable.

Thus, only about 20 accidents and incidents of significance have occurred in 246 assorted reactors of all kinds, operated for a cumulative total of 1 029 reactor years. No member of the general public has received over-exposure to radiation from reactor accidents. No accident has caused interference with normal activities of the general public. Only three reactor employees have been killed, all in one accident, and the number who have been over-exposed from the accidents *per se* (not including planned and controlled over-exposures in subsequent clean-up operations) is small. The cumulative record of experience with reactor operations and with reactor accidents gives increasing confidence that the probability of catastrophic accidents and the likelihood of substantial damage to public health and safety is indeed extremely low, particularly for large steadily-operated power reactors.

SUMMARY

To the present time, reactors in the United States have been so located as to place dependence for protection of people against the consequences of possible accidents in part on conservative design and engineered safeguards and in part on isolation distances from heavily populated areas. The accident record thus far in the 246 reactors of all assorted types which have been operated in the United States during the 21 year period since criticality of the first reactor, for a total of over 1 029 reactor years, has been extremely good.

As knowledge grows of the characteristics and predictable behavior of reactors, as confidence increases in the dependability of engineered safeguards, and as the record of successful and safe operation of reactors is extended, a basis is laid for increased reliance to be placed on design and engineered safeguards with a corresponding reduction in the isolation distance requirements. This trend is already observable in the sites approved for some of the latest reactors of well-proven design in the United States. Any extension of this trend will depend on the continuation of successful operating experience and on the evidence that can be assembled on the level of confidence to be placed on the dependability of engineered safeguards.

ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/275 Etats-Unis d'Amérique

Expérience des Etats-Unis en matière de réacteurs et implantation des réacteurs de puissance

par C. K. Beck

A la base, le seul danger exceptionnel pour la santé et la sécurité du public que présente le fonctionnement de réacteurs nucléaires provient de la possibilité pour des matières radioactives produites par les réacteurs au cours de leur fonctionnement de s'échapper d'une manière ou d'une autre et de se répandre en quantités dangereuses dans des régions à forte densité de population. Il s'ensuit que la sécurité du public peut être assurée si l'on peut trouver des moyens d'empêcher l'échappement de quantités dangereuses de substances radioactives dans des régions habitées. Les efforts

destinés à garantir cela portent sur deux domaines principaux:

a) Etude, construction et utilisation de réacteurs possédant des dispositifs de sécurité suffisants à la fois pour la prévention d'accidents et pour la minimisation des conséquences d'accidents, et

b) Implantation des réacteurs de telle manière que les distances assurant l'isolement ainsi que les autres caractéristiques du site soient convenables.

Le mémoire récapitule certains des principes de sécurité fondamentaux applicables à l'étude, la construction et l'utilisation de réacteurs de puissance et décrit les concepts sur lesquels reposent les critères de sélection de sites d'implantation de réacteurs. A propos du problème du choix d'un site, on discute les rôles joués par les « pires accidents possibles » et les « indices de dose d'exposition », la possibilité de réduire les distances imposées à mesure que la sûreté des dispositifs techniques de sécurité est améliorée, une liste préliminaire des principes liés à l'évaluation des divers dispositifs de sécurité, et une indication de la tendance actuelle en matière d'implantation de réacteurs aux Etats-Unis.

Jusqu'à présent, plus de 225 réacteurs de types divers et de différents niveaux de puissance ont été construits et mis en fonctionnement aux Etats-Unis... représentant un nombre cumulatif très élevé d'années de réacteurs et de mégawattjours d'expérience de fonctionnement. Les vingt réacteurs de puissance exploités sous licence (dont quatre sont encore au bas niveau de fonctionnement initial) totalisent à eux seuls plus de 1 250 000 mégawattjours pour les six années qui se sont écoulées depuis que le premier a été mis en service. Pas un seul accident de réacteur important (c'est-à-dire un accident ayant causé des dommages à des individus ou à l'installation) ne s'est produit.

Pour les quelque 200 autres réacteurs divers: de recherche, expérimentaux, de puissance (appartenant à l'Etat), de propulsion et de production, des chiffres beaucoup plus impressionnants (qui sont actuellement rassemblés pour la mise à jour des totaux) ont été enregistrés. Ces réacteurs ont été exploités pendant plus de vingt ans dans toutes sortes de programmes expérimentaux et d'essais ainsi que dans des conditions de fonctionnement stable, et à une époque où nos connaissances étaient beaucoup moins étendues qu'actuellement.

Malgré cela, durant cette période totale de fonctionnement, il s'est produit seulement 15 accidents de réacteurs ayant entraîné soit des dommages corporels (trois personnes ont été tuées, toutes trois dans le même accident), soit des dégâts matériels à l'installation, ou de grosses excursions de puissance fortuites (sans qu'il en résulte de dommages).

Le mémoire comprend une brève discussion de ces accidents et de leurs incidences en ce qui concerne la sûreté des réacteurs.

En appendice, le mémoire contient une description des règles et pratiques administratives actuelles en matière d'évaluation de la sûreté, de l'attribution de licences, de l'inspection et de la surveillance des réacteurs licenciés aux Etats-Unis.

A/275 США

Опыт США по размещению энергетических реакторов

К. К. Бек

Единственная потенциальная угроза для здоровья и безопасности населения, связанная с эксплуатацией ядерного реактора, обусловлена возможностью утечки радиоактивных материалов, образующихся в реакторе в ходе его эксплуатации, и проникновения их в населенные районы в опасных количествах. Отсюда следует, что общественная безопасность может быть обеспечена, если будут найдены средства, предотвращающие утечку опасных количеств радиоактивных материалов в населенные районы. Работы по созданию таких средств ведутся в двух основных направлениях:

а) проектирование, строительство и эксплуатация таких реакторов, защитные средства которых позволяют как предотвращать аварии, так и уменьшать последствия аварий до минимума;

b) осуществление такого выбора места для размещения реакторов, при котором изоляция и прочие характерные особенности участка являются удовлетворительными.

Настоящий доклад содержит описание основных принципов, которые могут быть использованы при проектировании, сооружении и эксплуатации безопасных энергетических реакторов, в нем также приводятся современные критерии выбора площадки для строительства реактора. Рассмотрение вопросов, связанных с размещением реакторов, включает обсуждение роли, отведенной «предельно возможным авариям» и «показателям доз облучения», а также анализ возможности снижения требований к величине изолирующего расстояния по мере совершенствования технических средств безопасности. Обсуждаются также метод оценки индивидуальных мер безопасности и современные тенденции в выборе площадок для строительства реакторов в Соединенных Штатах.

К настоящему времени в США построены и находятся в эксплуатации более 225 реакторов разнообразных типов и мощностей, обладающих, согласно проведенным расчетам, неожиданно большим общим числом накопленных лет рабочей жизни и мегаватт-суток эксплуатационного опыта.

Только у двадцати энергетических реакторов (четыре из которых до настоящего времени эксплуатируются на мощностях ниже номинальных) за шесть лет, прошедших со времени пуска первого из них, зарегистрированы более $1{,}25 \cdot 10^6$ *Мвт · сутки*. За это время не произошло ни одной значительной аварии (то есть такой, которая причинила бы значительный ущерб людям или установке).

У остальных двух с лишним сотен разнообразных реакторов: исследовательских, испытательных, энергетических (правительственных), для двигателей и производственных реакторов зарегистрирован гораздо более внушительный стаж работы (в настоящее время завершается его определение). Эти реакторы проработали более двадцати лет во всевозможных экспериментальных и испытательных программах, а также эксплуатировались в установившемся режиме, включая те времена, когда знания были гораздо менее обширны, чем теперь. Однако за все время эксплуатации реакторов произошло только 15 аварий, в ходе которых были нанесены повреждения людям (три человека было убито во время одной аварии), ущерб установке или имели место неожиданные скачки мощности большой величины (не сопровождавшиеся повреждениями).

Кратко рассматриваются причины этих аварий и анализируется их влияние на повышение безопасности реакторов.

В виде приложения доклад содержит описание существующего в настоящее время порядка оценки безопасности реакторов, выдачи лицензий, инспекции и наблюдения за реакторами, на которое имеется разрешение на работу в Соединенных Штатах.

A/275 Estados Unidos de América

Experiencia de los Estados Unidos en relación con la seguridad y el emplazamiento de reactores de potencia

por C. K. Beck

Básicamente, la única amenaza que la operación de los reactores nucleares presenta para la salud y seguridad pública, tiene su origen en la posibilidad de que los materiales radiactivos generados por el reactor durante su operación puedan escapar de alguna manera y concentrarse en cantidades nocivas en zonas pobladas. Por consiguiente, la seguridad pública puede asegurarse si se encuentran los medios de prevenir el escape de cantidades nocivas de materiales radiactivos a las zonas públicas. Los esfuerzos para alcanzar este fin se concentran en dos cuestiones principales:

a) Diseño, construcción y operación de los reactores de manera que posean características adecuadas para la prevención de accidentes y para reducir a un mínimo las consecuencias de los accidentes, y

b) Situación de los reactores de modo que su aislamiento y otras características del lugar de instalación sean apropiadas.

La memoria resume algunos de los principios básicos de seguridad aplicables al diseño, construcción y operación de reactores de potencia y describe los conceptos en que se basan las presentes guías para la selección del emplazamiento del reactor. Las consideraciones sobre el lugar de posible instalación de un reactor incluyen una discusión de la importancia del « accidente máximo probable » y de « los índices de exposición », la posibilidad de reducir las distancias requeridas a medida que se mejora el diseño de los métodos de seguridad, una lista preliminar de principios relacionados con diferentes procedimientos de seguridad y una breve mención de las tendencias actuales en lo que respecta a la localización de reactores en los Estados Unidos.

Hasta el presente, más de 225 reactores de diferentes tipos y niveles de potencia han sido construidos y operados en los Estados Unidos... lo que ha dado como resultado una experiencia equivalente a la adquirida después de operar reactores durante muchos años, produciendo energía equivalente a un gran número de megavatios-día. Solamente con los 20 reactores de potencia autorizados (cuatro de los cuales están todavía operando a los bajos niveles de potencia que se requieren en los primeros tiempos de operación), más de 1,25 millones de megavatios-día de operación han sido registrados en los seis años que han transcurrido desde que el primer reactor comenzó a operar. Ni siquiera ha ocurrido un solo accidente importante (es decir, que haya dañado la instalación o causado daños personales).

De los otros 200 reactores de diferentes tipos: investigación, pruebas, potencia (del gobierno), propulsión y producción, se ha hecho un registro de operación mucho más impresionante (que ahora se está recopilando para conocer las cifras totales actuales). Estos reactores han sido operados durante más de 20 años en muchos tipos de programas de prueba y de experimentación, como también en condiciones estacionarias, y en momentos en que nuestros conocimientos eran mucho menos extensos de lo que son ahora. Sin embargo, durante estas operaciones, solamente ha habido 15 accidentes que hayan causado daños personales (tres personas resultaron muertas, todas en un mismo accidente), daños a la instalación o grandes aumentos inesperados del nivel de la potencia (sin daños personales).

Se presenta una breve discusión de estos accidentes y su relación con la seguridad en la operación de reactores.

En apéndice a la memoria figura una descripción de los procedimientos administrativos y prácticos relacionados con la evaluación de la seguridad, autorización de operación, inspección y vigilancia de los reactores autorizados de los Estados Unidos.

Containment and engineered safety of nuclear power plants

By W. K. Davis * and W. B. Cottrell **

All nuclear reactors depend on inherent safety characteristics, added engineered safeguards, and administrative controls to prevent accidents which could lead to release of radioactivity and to limit the consequences of such an accident. To the extent that these factors cannot provide a sufficient degree of protection, remote siting is also used to reduce the consequences of a major release of radioactivity. However, remote siting imposes substantial cost penalties on nuclear power plants, a fact of concern with nuclear power becoming economically attractive in many areas.

Economic and suitable sites for any type of power plant are becoming scarce in many areas. Additional limitations resulting from nuclear hazards consideration may seriously restrict the use of nuclear power. Great emphasis is, therefore, placed on developing means by which nuclear power plants can be located at sites otherwise suitable for power stations. This has not been a controlling consideration for research, testing, or prototype power reactors. It is believed suitable means can be developed to effectively and reliably reduce the consequences of an accident in a nuclear plant to acceptable levels at almost any location and thus allow nuclear power to achieve its full economic potential.

Considerable effort has been devoted to demonstrating the inherent safety characteristics of reactors, and it is now generally accepted that the probability of a major release of radioactivity from a power reactor system is very remote. However, in spite of the inherent characteristics and added engineered safeguards which minimize the possibility of an accidental release of radioactivity from the reactor system, it does not appear possible to completely preclude such incidents, however unlikely they are considered to be. Thus, attention has been directed to limiting the consequences, especially to the public, of abnormal releases of radioactivity from the reactor and all US nuclear power plants have been designed to cope with such releases. It is principally in this area of limiting the consequences of accidents that the means of reducing siting limitations for large nuclear power plants are being developed.

This paper reviews the containment systems and other engineered safeguards features used or proposed for use with power reactors of US design, with emphasis on light-water cooled reactors. "Containment" is taken to mean all structures, systems and devices provided to confine radioactive materials which have been released inadvertently from the reactor system. The term "engineered safeguards" generally refers to all features of the plant design which are provided to enhance the safety of the plant, including accident-preventing features as well as containment and other consequences-limiting features. "Auxiliary safeguards" are defined more specifically as those consequences-limiting features, in addition to the basic containment system, which may be incorporated into the plant design to further reduce the hazard to the public of a serious accident in the reactor system.

CONTAINMENT SYSTEMS

The principal requirements and design considerations for reactor containment were discussed at the Second International Conference in 1958 and containment designs for several plants were described [1-4]. Since that time several nuclear power plants have been built in the United States, most of them using steel shell pressure containment of the type described most frequently at the 1958 Conference. This type of structure is well-suited to meet the structural and leakage requirements of power reactor containment and will continue to be widely used. However, other containment concepts have been developed and are being used or proposed for various applications where they may have significant advantages in cost or effectiveness over steel pressure containment vessels.

The principal types of containment now being considered in the United States are illustrated schematically in Figs. 1 through 6. Although most of these containment types are similar in many respects and any categorizing must be somewhat arbitrary, they

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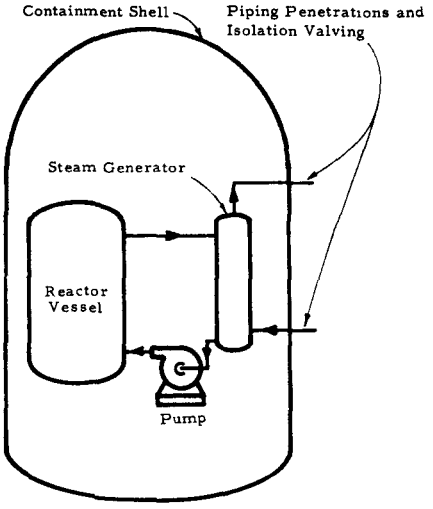


Figure 1. Pressure containment (shown with PWR, GCR or OCR)

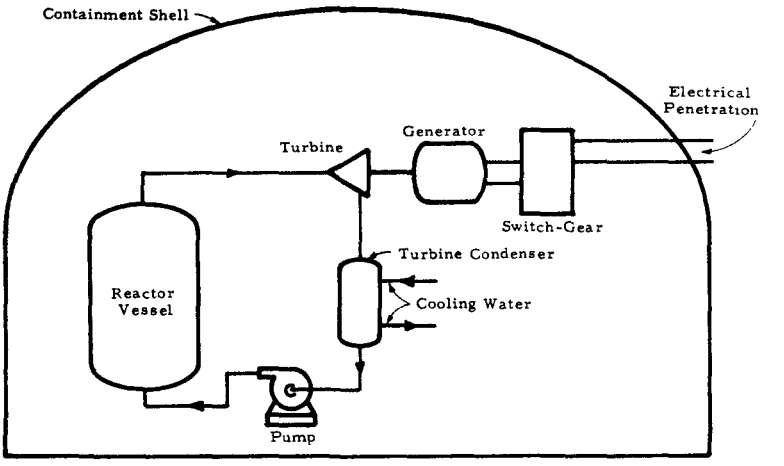


Figure 2. Low pressure containment (shown with BWR)

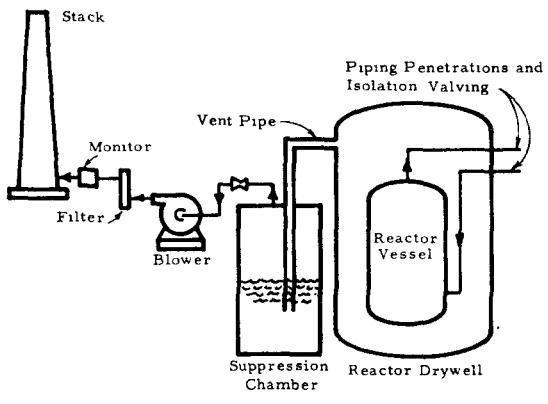


Figure 3. Pressure suppression containment (shown with BWR)

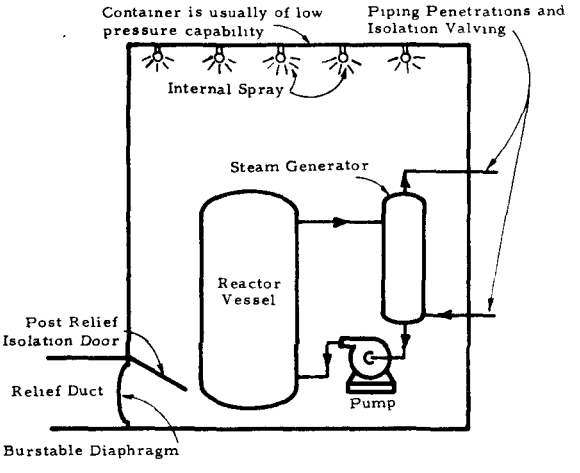


Figure 4. Pressure relief containment (shown with PWR)

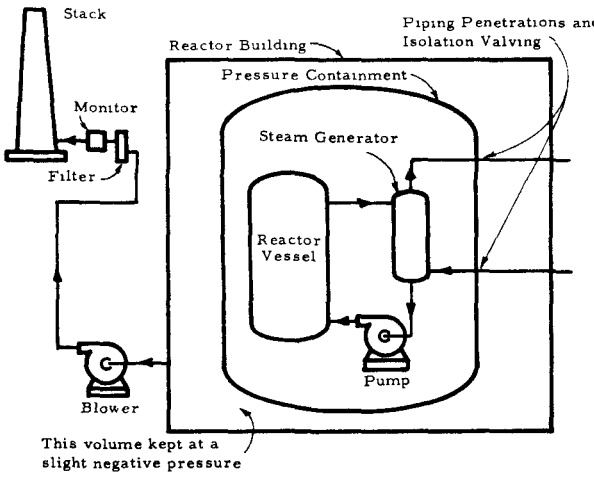


Figure 5. Multiple barrier containment (shown with PWR)

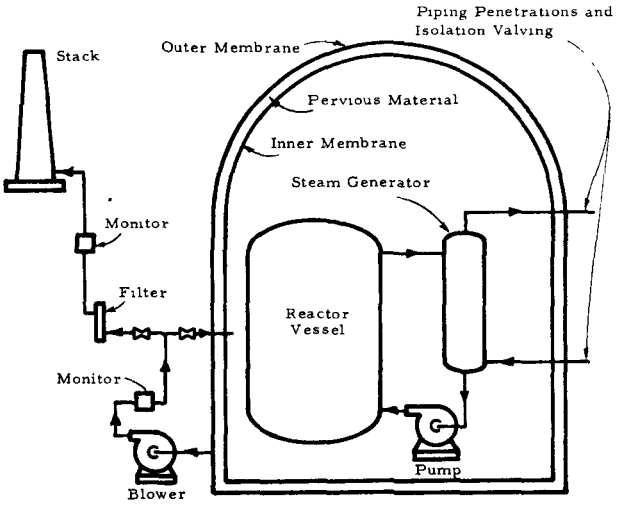


Figure 6. Multiple barrier containment (shown with PWR)

are described according to their principal distinguishing features.

Table 1 lists the principal characteristics of the containment systems used on several United States power reactors. The Reactor Containment Handbook [5] contains more information on these and other containment designs.

Pressure containment

The large majority of nuclear power plants built in the United States have been provided with pressure-containing steel shell containment vessels. This type of containment is designed to contain all the material, including any fission products, released from the reactor system in the event of an accident and consequently is designed as a pressure vessel in accordance with applicable pressure vessel codes. Spheres and cylinders with ellipsoidal and hemispherical ends have been used.

Allowable leakage rates as low as 0.1% per day have been specified for many welded steel containment vessels and initial leakage tests have indicated leakage rates below this value with estimated errors about equal to the leakage rate. While it is believed that leakage rates substantially less than 0.1% per day can be achieved by careful design, construction, testing, and maintenance of the vessel and the penetrations, the measurement of such leak rates on an integral basis will be extremely difficult because of changes in temperature, humidity, atmospheric pressure, etc.

Reinforced concrete as well as steel plate can be used for pressure containment although it has little use to date. If containment shielding is required, a single concrete pressure vessel may perform both the shielding and containment functions more economically than a steel vessel with a separate concrete shield. A welded steel liner is usually required to provide the required degree of leak tightness.

A conventional pressure containment system is illustrated schematically in Fig. 1.

Low pressure containment

Low pressure containment is defined as a containment structure with sufficient volume that the design pressure is below some arbitrary value, normally about 5 psig. Maintaining the containment design pressure below 5 psig requires either that a relatively small amount of energy be released from the reactor system during an accident or that a very large volume is provided within the containment system. Thus the concept is most suitable for other than water-cooled reactors or for low power, water-cooled reactors. Low pressure containment is also appropriate for those plants in which it is considered desirable to contain the entire system including the turbine-generator. This type of containment has been referred to as "total containment". The containment of the BONUS plant in Puerto Rico is of this type, consisting of a large hemispherical steel dome mounted on a thick concrete slab [6]. Although this reactor produces only 50 MW(th) of power, the containment system has a free volume greater than 1.5 million cubic feet and a design pressure of 5 psig.

A low pressure containment concept, similar to that used for BONUS, is illustrated in Fig. 2.

Pressure suppression containment

Pressure suppression is a containment concept based on absorbing a large part of the energy released from the reactor system as it is released and thereby reducing the design pressure (or required free volume) of the containment structure. This concept, developed primarily for use with water-cooled reactors, employs a large pool of water to condense the steam which would be released in the event of a primary system rupture. The pool of water also retains a significant portion of the fission products released from the reactor. This concept is more applicable to single cycle boiling-water reactors than to pressurized-water reactors because the relatively smaller amount of coolant requires a smaller volume of water to condense the steam and because the smaller containment volume required to envelope only the reactor system permits greater advantage to be taken of such a containment concept which requires less volume to contain the material released. A principal design requirement of pressure suppression containment is to ensure that adequate venting capacity is provided from the vessel surrounding the reactor (the drywell) to the suppression pool.

The principal features of pressure suppression containment are illustrated schematically in Fig. 3.

Additional advantages can be claimed for the pressure suppression concept if the various components of the system are suitably arranged. The compact design of the entire system makes it suitable for underground use as in the case of the Humboldt Bay plant [7,8]. In the Humboldt Bay design, a relatively small drywell surrounds the reactor vessel which, in this natural circulation, internal steam separation reactor design, is essentially the entire primary system. Vent pipes connect the drywell with the pool of water in the suppression chamber, an annular space surrounding the drywell. In this particular plant, in which the nuclear unit was added immediately adjacent to two existing oil-fired units, complete containment shielding and a compact plant design were required. These requirements, together with the local soil conditions, resulted in this design using pressure suppression and a caisson construction technique.

Table 1. Containment characteristics of several nuclear plants [a]

Reactor	Reactor type	Reactor power MW(th)	Containment type and shape	Free volume 10^3 ft^3	Design pressure	Leakage requirement	Basis for design	Actual site distances: Exclusion radius miles	Actual site distances: Population center dist., miles
Big Rock Point	Boiling water	157	Steel sphere partially below grade	922	27(-1.22) psig	0.5%/day	Loss of coolant through a double-ended failure of the largest pipe, when in hot standby	0.50	135.0
CVTR	Pressurized tube	63	Reinforced concrete cylinder partially below grade with steel dome top covered with concrete	243	21 psig	0.1%/day	Loss of primary coolant followed by 35% Zr-water reaction and 10^6 Btu nuclear reaction. Core injection system is assumed to fail	0.50	25.0
Dresden	Boiling water	626	Steel sphere partially below grade	2 880	29.5 (-1.0) psig	0.5%/day	Loss of all primary coolant instantly; 25% of the Zr reacts with H_2O; nuclear excursion to melt fuel	0.50	14.0
Elk River	Boiling water with nuclear superheater	72.7 (58.2+ 14.5)	Vertical steel cylinder with hemispherical top and hemillipsoidal bottom, partially below grade	287	21(-0.33) psig	0.1%/day	Loss of coolant through largest possible pipe rupture (10 s emptying time); no heat loss	0.23	20.0
Indian Point	Pressurized water thorium converter	585	Steel sphere partially below grade enclosed in concrete shield	1 800	25(-1.25) psig	0.1%/day	Instantaneous release and expansion of all primary fluid and secondary fluid from one steam generator	0.30	17.0
VBWR	Boiling water	30	Vertical steel cylinder with hemispherical ends, partially below grade	125	45(-2) psig	1%/day	Rupture of reactor pressure vessel; pressure vessel contents released instantly, remainder of coolant in 1.5 s	0.40	15.0
Yankee	Pressurized water	485	Steel sphere above grade	860	34.5 psig	0.1%/day	Complete loss of primary coolant through 20-in (largest) pipe break; core injection prevents fission product release (hypothetical accident same but core injection assumed to fail and fission products released)	0.50	21.0
BONUS	Boiling water with nuclear superheater	50	Low pressure vertical steel cylinder with hemispherical steel dome and flat concrete bottom	1 590	5(-0.25) psig	0.2%/day at 5 psig	Total loss of primary coolant through largest credible break (16-in line) at bottom of reactor; pressure vessel empties in 4 s. Core injection system assumed to fail	0.3	
Humboldt Bay	Boiling water (natural circulation)	165	Pressure suppression:				Loss of primary coolant through largest credible rupture within the drywell	0.25	3.0
			Drywell: vertical cylinder; concrete lined with steel	12.5	72(-0.5) psig	0.1%/day at 72 psig			
			Supression chamber: vertical cylindrical annulus; concrete lined with steel	33.4 (above water)	10 psig	1%/day at 10 psig			
			Refueling building: negative pressure; parallel-piped; reinforced concrete	192 (includes pipe tunnel)	± 7 in H_2O	100%/day at -1/4 in H_2O	One dropped fuel rod		
Hallam	Sodium cooled	256	Refueling building: negative pressure; rectangular, mill-type building	1 420	-1/4 in H_2O	3 120 ft^3/min at-1/8 in H_2O (inleakage)	One dropped fuel element, with subsequent oxidation of that element	0.25	17.0

[a] All data except last two columns taken from Chapter 8 of Ref. [5].

In the proposed Bodega Bay pressure suppression containment design [9], the suppression chamber is a torus-shaped steel pressure vessel enclosed in a concrete vault. In addition to the containment boundaries provided by the drywell, the suppression pool, and the wall of the suppression chamber, any leakage can be collected in the vault and vented through a filter system to the ventilation stack. In addition, the Bodega Bay design provides space around the primary steam line and the turbine-generator which can be vented through the filter system to the ventilation stack so any leakage from the steam system of this direct cycle reactor plant can be collected and controlled.

Pressure relief containment

The pressure relief containment concept also reduces the total amount of material which must be contained by the containment structure, in this case by venting the initial pressure surge to the atmosphere. The vent is then closed and the containment enclosure sealed before fission products are released from the reactor. This concept is based on the assumption that little radioactivity will be released until the coolant has boiled off, the fuel has overheated and the cladding has ruptured, a process requiring an appreciable time. This concept has been used on the Canadian NPD reactor [10] and on the NPR plant at Hanford [11]. In both of these designs, a spray or fog system is used to prevent a pressure build-up in the containment enclosure due to decay heat after the initial pressure surge has been vented and the vent closed. This concept is illustrated in Fig. 4.

A principal design requirement of this and other types of rate-dependent containment (e.g., pressure suppression) is to provide sufficient relief capacity so the maximum release from the primary system can be relieved fast enough to prevent a high peak pressure. It is also imperative that reliable vent closures be provided to seal the vent system after the initial pressure surge and before the radioactivity can be released from the reactor. Multiple or back-up closure devices are necessary. If the peak accident pressure in such a containment system can be kept to very low levels, low leakage buildings may be used rather than pressure vessel design and construction techniques. Leakage rate specifications similar to those for other types of containment at similar locations must be met.

Negative pressure containment

Buildings utilizing rather conventional building construction with special provisions to reduce leakage may be used for containment where the maximum assumed accident will not release large amounts of material and the pressure to be contained is negligible. The most common application of such types of buildings for containment is the so-called "negative pressure containment" concept. In this concept, the ventilation system is provided with sufficient capacity to maintain a slightly reduced pressure within the building so all leakage through the building walls is inward during any anticipated barometric conditions and under the maximum assumed accident conditions. The ventilation system exhausts through a system of filters to a stack. In this way, all radioactivity released to the building can be controlled and cleaned up. The method depends on sufficient blower and filter capacity and on continued operation of the blowers and filters after an accident has occurred.

The reactor building shown schematically in Fig. 5 is a negative pressure containment structure, in this case used in conjunction with a pressure containment vessel to form a multiple barrier containment system as discussed below.

The Oak Ridge research reactor containment is of this type [12]. The Hallam [13] and Humboldt Bay [8] refueling buildings also employ this concept to provide containment for an assumed refueling accident. However, this concept has not been used as the primary containment for any nuclear power plants in the United States.

Multiple barrier containment

Multiple barrier containment, typical illustrations of which are shown in Figs. 5 and 6, offers a greater degree of leakage control than any containment concept yet proposed. In this concept, all leakage through the first barrier is collected within the reduced pressure zone formed by the second barrier and is exhausted through a filter system and up a stack. In the design shown in Fig. 6, the leakage collected in the intermediate space also can be pumped back into the first vessel. Even if the blowers which remove the air from the intermediate space should fail, there would be a considerable time delay before leakage out of the first barrier could build up sufficient pressure to cause leakage out of the second barrier and a substantial advantage can be obtained from the resulting fission-product decay.

The first barrier in most containment systems of this type is a conventional steel shell containment vessel with a low leakage rate. The second barrier can be a low leakage building or other structure, as in the case of the NS *Savannah* [14] and as shown in Fig. 5; it could be a concrete containment shield, as in the case of the Indian Point plant [15]; or it may be a second high integrity steel shell as shown in Fig. 6 and as proposed for the City of Los Angeles plant at Malibu and for the Ravenswood plant [16]. In order to employ the pump-back concept, the second barrier must be essentially leak tight to prevent too rapid a

rise in pressure within the primary containment boundary. To take full advantage of the multiple containment concept, all penetrations should be doubly contained with the intermediate space vented to the space between the two containment boundaries.

Underground containment

Underground reactor location automatically provides shielding of the containment. Protection is also afforded from external missiles or aircraft, but this has not been a consideration in the US. There may also be psychological advantages and one of æsthetics in some locations.

In order to obtain the necessary leak tightness, a liner or internal structure will almost always be required. Steel or reinforced concrete structures may be needed as well to provide the necessary pressure-containing capability. Penetrations remain a problem of the same magnitude as for surface construction. The cost of excavation, particularly in the US with high labor costs, are high [17] and perhaps more important are the unknowns from unforeseen obstructions, poor rock, and water. Access and installation of equipment will also be expensive.

Underground containment has been used to some extent in Europe [18, 19, 20] but not in the United States.

AUXILIARY SAFEGUARDS

Various auxiliary safeguards systems and devices are included in most nuclear plant designs to augment the basic containment system and improve its effectiveness in reducing the consequences of an accident in the reactor system. These added engineered safeguards may be integral parts of the containment system but since they are identifiable features and may result in significant differences in effectiveness between containment systems of similar basic design, they may be discussed separately from the containment system itself.

It is not possible to make a general statement concerning the extent to which auxiliary safeguards are required in the containment systems of power reactors or the extent to which they can be utilized to substitute for other safety aspects of the containment design or of the plant site. It is apparent, however, that reliance on some additional containment and auxiliary safeguards features will be required for siting of large nuclear power plants in populated areas where conventional containment systems are not considered adequate. The regulating authorities in the United States have demonstrated their willingness to consider additional engineered safeguards as at least a partial substitute for remoteness, provided these safeguards can be demonstrated to be effective and reliable. As yet few precedents have been established which indicate the degree to which a particular type of safeguard might be considered effective in reducing site distance requirements. Nor is it likely firm criteria can be established since a particular safeguard feature can only be considered as it applies to a particular containment design and to a particular reactor design at a specific site. This becomes a matter of judgement involving consideration of all factors affecting the hazards of the plant.

The following paragraphs discuss briefly some of the auxiliary safeguards systems and devices used or proposed for reducing the consequences of an accident in the reactor system. Further details are given in the Reactor Containment Handbook [5] chapter 8. Table 2 summarizes some of the safeguards features used on several nuclear plants in the United States.

Core spray and injection

Core spray or injection systems are included in many water-cooled reactor plant designs to prevent the fuel from overheating and releasing fission products if the primary coolant is lost. These systems either pump water into the primary system where it can flow past or immerse the core, or they spray water directly onto the fuel surfaces. Large volumes of water are required, usually from a source external to the plant. Recirculating pumps may be provided to permit re-use of the water and to prevent the containment vessel from becoming filled with water and overstressed.

If the core cooling system functions properly, release of fission products is prevented or kept to a small fraction of what it would otherwise be. However, because this system must be connected to or closely adjacent to the primary system and thus faces the possibility of damage in an accident which ruptures the primary system, it is sometimes assumed to be inoperable for purposes of designing the containment and other safeguards systems. This does not mean such systems should not be used, however, since it is worth while to reduce the probability of releasing fission products from the reactor system.

Building sprays and fogs

Water spray and fog systems are used in several containment concepts, both to condense steam and reduce the pressure resulting from an accident, and to wash fission products out of the containment atmosphere in order to reduce the direct radiation from the containment structure and to reduce the inventory of fission products available for leakage out of the containment. Spray systems are employed principally for pressure reduction whereas fog may be more effective in removing fission products from air

Table 2. Engineered safeguards features of several nuclear plants [a]

Reactor	Core spray or injection			Building spray			Filter systems			
	Rate	Arrangement	Comment	Rate	Arrangement	Comment	Rate	Function	Description	Comment
Big Rock Point	1 000 gpm maximum addition rate; 400 gpm recirculation rate	Spray ring in reactor vessel. Spray actuated by low primary system pressure	Cooling limits core melting	1 000 gpm maximum addition rate; 400 gpm recirculation rate	Two sets of spray nozzles: one automatically actuates when sphere pressure reaches 2 psig; other operated manually from outside sphere. 100% coverage	Spray systems furnish pressure reduction after accident; can be operated from a remote, shielded location				
CVTR	1 500 gpm at 2 185 ft of D_2O	Injection into inlet headers (secondary injection into primary pump suction)	No credit taken at MCA				1 000 ft³/min	Recirculation for particulate and iodine removal	Prefilter, absolute filter (99.9% removal of 0.3—μ particles) and iodine removal section of activated charcoal (1 in deep, 90% iodine removal). Two such units, one in standby	No credit taken at MCA
Elk River	12 gpm minimum for 500 min	Spray ring in reactor vessel	Core spray depended upon to limit extent of core melting	1 000 gpm	Spray headers located above main and basement floors give 100% coverage	Spray reduces pressure after accident	3 500 ft³/min (max.)	Once through for particulate removal	Prefilter, absolute filter (99.95% removal of 0.3—μ particles)	
Indian Point				Internal: 1 000 gpm	Eight spray headers in top hemisphere located to wash down surfaces	No credit taken for either internal or external spray cooling at MCA	10 000 ft³/min	Once through for particulate removal	Prefilters; airborne activity filters (99.97% removal of 0.3—μ particles)	No credit taken at MCA
				External: 3 000 gpm	Eight recirculating spray headers in top hemisphere					
Yankee	1 800 gpm	Injection of borated water into each of four main coolant loops; actuates when primary system pressure drops to 1 000 psi	No credit taken at MCA				12 000 ft³/min (3 units each of 4 000 ft³/min capacity)	Recirculates containment air to remove particulates	Prefilters; airborne activity filters (99.95% removal of 0.3—μ particles)	No credit taken at MCA
BONUS	100 gpm (maximum)	20 nozzles direct spray, 60% to fuel, 40% to superheater elements	No credit taken at MCA	Internal: 1 000 gpm	Headers and nozzles over main floor and in basement	No credit taken at MCA for either system; both can be operated from remote location	50 000 ft³/min (bldg. ventilation)	Once through for particulate removal	Oil-treated prefilter and glass fibre filter	
				External: 450 gpm	Two circular perforated ring headers around upper dome					
Humboldt Bay	350 gpm at 140 psi	Spray ring near top of pressure vessel. Spray actuates at primary system pressure of 150 psig	No credit taken at MCA				Refueling 134 ft³/min	Building: Once through for particulates and iodine removal	Caustic scrubber and absolute filters (95% minimum removal of halogen and particulates)	

[a] From Chapter 8 of Ref. [5].

since the water droplets are smaller and can interact better with very small particles of fission products. Relatively little quantitative information is available on the use of spray or fog systems fog removing fission products from a containment atmosphere, and conservative estimates of the effectiveness of these methods must now be used in designing and evaluating such systems. In some plants where an internal spray system is provided, no credit is taken for fission-product removal although it certainly will occur to some extent. Spray systems can only be used for those types of reactors in which the addition of water will not cause additional problems. Reliability of a spray system may be improved by utilizing a supply of water contained in an elevated tank within the containment system so the sprays may be gravity fed and not require an external power supply. However, the supply of water which can be provided in this manner is limited and some means for additional water supply or recirculation is required for larger power plants.

Filter systems

Filter systems of various types are normally used on ventilation stacks to remove radioactivity from ventilation air exhausted to the atmosphere. In negative pressure containment systems, which depend on filter systems for removing radioactivity from a large volume of air during an accident, such filter systems may be quite large and elaborate. On conventional containment systems, which are isolated rather than ventilated in the event of an accident, the amount of radioactivity which must be removed by the ventilation system filters is quite small and dependence upon them is not so great.

Filters also may be used in recirculating systems within the containment system to reduce the inventory of fission products within the containment atmosphere. The filters may be shielded so that the fission products collected do not become an external radiation source. It also may be necessary to provide cooling to the filter elements so the decay heat of the trapped fission products does not cause damage to the filters.

In both applications of filter systems, a high efficiency filter preceded by a prefilter or roughing filter is normally used. In cases in which the filter system must operate in a high humidity environment, a moisture separator may be required ahead of the filter.

Iodine removal systems

Various types of iodine removal systems are used in conjunction with particulate filters to remove radioactive iodine both from the ventilation exhaust of a negative pressure containment system and from a recirculating air stream within a containment vessel after an accident. Since iodine usually is the limiting radioactive material released during an accident, iodine removal is particularly important. Activated charcoal, caustic scrubbers and silver-plated wire mesh are the most commonly used iodine removal devices. As with particulate filters, care must be taken in designing these systems so the accident conditions, notably high humidity and high temperature, will not render these devices ineffective when most needed.

Foams

The use of stable foams for trapping fission products within a containment system has been proposed but as yet has not been thoroughly tested or applied to a specific containment design.

Additional containment accessories

Other features which may be considered in the category of auxiliary safeguards include double and quick-acting isolation valves, double electrical and piping penetrations, air locks, and multiple instrumentation and control channels. Containment effectiveness and reliability may be improved by providing additional or back-up features of this type. In some cases these multiple features facilitate control of leakage or leakage testing. In other cases they may improve reliability by increasing the chances that at least one of several devices will operate when it is required.

RELIABILITY

As containment and other engineered safeguards systems become increasingly important in the design and siting of nuclear power plants, greater attention is being directed to the question of the reliability of these systems. A system which can be shown to be highly effective under carefully controlled conditions is of little value if it cannot be counted upon to function when it is required.

In designing engineered safeguards for maximum reliability, several factors should be kept in mind. Static systems and devices should be used wherever possible rather than those which require electrical or mechanical action in order to function properly. Inherent mechanisms for mitigating an accident should be utilized and enhanced to the maximum extent. Mechanical devices should be designed to fail in safe position so protection will not be lost upon loss of power or other malfunction. They also should be designed to operate after long periods of inactivity. And, an obvious but sometimes overlooked point, safeguards systems must be designed to operate under accident conditions, not just under idealized design and test conditions.

For some critical systems upon which the entire operation of the containment system depends, multiple or redundant systems may be required. This is particularly true for instrumentation which senses that an accident has occurred and provides an alarm or initiates protective action. A highly reliable mechanical device is of no use if the sensors, circuiting, and power supply which cause it to function are unreliable. Spurious signals which cause a safety system to actuate in error could cause severe operating difficulties. For these reasons, several independent sensors, sometimes operating in coincidence, may be used to automatically actuate a safeguards device or provide an alarm. Two complete and independent containment closure devices and their associated control systems may be used in certain critical areas such as for isolation of an open containment ventilation system.

Since containment systems and other engineered safeguards are not normally called upon to function, their ability to function properly is not checked by normal operation. It is, therefore, particularly important that careful inspection, testing and maintenance of these safeguards systems be carried out to provide assurance of their continued operability and effectiveness. This implies that adequate administrative procedures for performing these functions must be available and that the necessary control is exercised to see that these procedures are followed. It is even more important that proper action be taken during abnormal situations for which there are no written procedures or when there is insufficient time to consult written procedures. This reliance on the human element, which may be termed "administrative safety", is, therefore, at least as important to the over-all reliability of the safeguards systems as is mechanical reliability. It requires alert and effective management, experienced and competent engineers and operators, and an over-all feeling or responsibility for safety by the designers, manufacturers and constructors, operators and maintainers, and regulators. Without this, any mechanical design can be rendered useless and any established procedure can be ignored.

Some things done in the name of technical safety may actually mitigate against administrative safety. The very fact that safety feature is added to safety feature may lead to an undue confidence that the system will operate as designed without due thought and attention. The very complexity of the system may make it difficult to analyze for basic defects or to understand it well enough so intelligent and proper action can be taken in case of an unanticipated failure or emergency.

It should be apparent that a proper balance between mechanical and administrative reliability must be achieved to avoid sacrificing one in seeking the other. The danger in excessive dependence on administrative controls lies in the fact that men can make mistakes — a well-recognized principle in reactor safety system design. The danger in overemphasizing the reliability of devices rather than men is that the sense of responsibility for the safe design and operation of the plant may tend to be taken from the designers and operators who, in the final analysis, determine how safe a plant will be.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/276 Etats-Unis d'Amérique

Enveloppes et dispositifs de sécurité pour centrales nucléaires

par W. K. Davis et W. B. Cottrell

Le mémoire passe en revue les dispositifs de sécurité utilisés dans les réacteurs de centrales électriques en fonctionnement ou à l'étude aux Etats-Unis. Lorsqu'elle existe, l'expérience pratique obtenue pour certains dispositifs de sécurité est présentée. De plus, des concepts nouveaux proposés pour satisfaire à des exigences d'implantation plus strictes sont exposés. Le mémoire ne traite pas des réacteurs de recherche, expérimentaux ou de propulsion.

Dans presque tous les réacteurs de puissance américains, le combustible utilisé est sous forme solide, généralement un matériau céramique de grande densité à point de fusion élevé enrobé d'une gaine à haute intégrité qui résiste à l'action corrosive du réfrigérant et retient tous les produits de fission. Le système de refroidissement est généralement à circuit fermé ou peut être fermé pour constituer un système à haute intégrité. Le réacteur et les parties du système de refroidissement primaire qui peuvent devenir radioactives sont entourés d'un bouclier en béton massif. De plus, dans tous les cas il existe une enveloppe de sécurité destinée à retenir toute matière qui pourrait s'échapper hors du système de refroidissement primaire et de son bouclier de protection.

Des produits de fission ne peuvent arriver jusqu'à l'enveloppe de sécurité qu'en conséquence d'un incident conduisant à la libération de produits de fission provenant du combustible (haute température) et la rupture du gainage et du système de refroidissement primaire. De plus, il existe des possibilités d'accidents au cours du chargement du combustible; elles sont considérées séparément. Deux genres d'incidents sont généralement envisagés: *a)* une excursion nucléaire assez importante pour provoquer la rupture du gainage et du système de refroidissement primaire, et *b)* une rupture du système de refroidissement primaire conduisant à la perte du réfrigérant et à la fusion du cœur, non refroidi, du réacteur. Ces deux types d'accidents sont considérés comme très improbables. Le mémoire présente des considérations concernant ces types d'accidents ainsi que les accidents lors du chargement du combustible dans le cas de divers réacteurs et une récapitulation des particularités inhérentes (coefficients de température, effet Doppler, etc.) et des dispositifs (systèmes de contrôle, systèmes de refroidissement de secours, etc.) utilisés dans ces réacteurs.

Les systèmes prévus pour limiter les conséquences de la libération de produits de fission, aussi improbable qu'elle puisse être, utilisent le confinement ou la rétention de toute matière libérée grâce à des cellules scellées de types divers, des enceintes d'acier ou de béton, ou des bâtiments permettant peu de fuites. Il peut exister deux barrières de ce genre. On peut prévoir des dispositifs destinés à limiter ou réduire le volume contenu (ou la pression) par suppression de vapeur, abaissement de pression, arroseurs, etc. De plus, divers types de systèmes de filtrage et d'arrosage peuvent être utilisés pour réduire la concentration de matières radioactives dans les fuites éventuelles hors de l'enceinte. Diverses méthodes destinées à réduire encore davantage les fuites hors de l'enceinte, telles que le repompage à partir d'une barrière secondaire, peuvent être utilisées. Le système de rétention doit être également protégé contre des projectiles et doit être éprouvé (et subir de nouvelles épreuves périodiquement) pour garantir l'étanchéité. Ces caractéristiques sont exposées et discutées pour les principaux types d'enveloppes de sécurité.

A/276 США

Безопасность работы оболочек и оборудования ядерных энергетических установок

У. К. Дэвис, У. Б. Котрелл

Настоящий доклад содержит описание мер, направленных на обеспечение безопасности энергетических реакторов США, как находящихся в эксплуатации, так и в стадии проектирования. Рассматривается опыт применения некоторых технических мер безопасности. Кроме того, обсуждаются новые критерии, предлагае-

мые для оценки площадок для строительства реактора. В докладе не описаны реакторы для научно-исследовательских и опытных работ или для двигателей.

Почти во всех конструкциях американских энергетических реакторов топливо находится в твердом виде, обычно это тугоплавкий керамический материал высокой плотности, заключенный в оболочку, выдерживающую коррозионное воздействие теплоносителя и удерживающую продукты деления. Обычно используются замкнутые системы охлаждения или такие, которые могут быть замкнуты для повышения их надежности. Реактор и наиболее радиоактивные части системы первичного теплоносителя окружены массивной бетонной защитой. Кроме того, всегда имеется оболочка, позволяющая предотвратить распространение любого материала, утечка которого возможна из системы первичного теплоносителя и ее защиты.

Продукты деления могут попадать в оболочку в результате аварии, в процессе которой создаются условия, приводящие к выделению продуктов деления из топлива (высокая температура) и разрушению оболочек тепловыделяющих элементов и системы первичного теплоносителя. Кроме того, существует вероятность аварий при перегрузке топлива, которая будет рассматриваться отдельно. Обычно рассматривается два типа аварий: 1) аварии, в процессе которых имеет место отклонение режима реактора, достаточное для того, чтобы прорвать оболочки и систему первичного теплоносителя, и 2) аварии, в процессе которых имеет место прорыв системы первичного теплоносителя, приводящий к потере теплоносителя и расплавлению неохлажденной активной зоны реактора. И тот и другой тип аварий считается маловероятным. В докладе рассматриваются вероятные последствия как указанных аварий, так и тех, которые могут иметь место при перегрузке топлива, для различных реакторов с учетом присущих им характеристик (температурных коэффициентов, эффекта Доплера и т. д.) и параметров оборудования (систем управления, аварийных систем охлаждения и т. д.), применяемого в этих реакторах.

Системы, используемые для предотвращения утечки продуктов деления, хотя и маловероятной, являются герметичными оболочками различного типа, изготовленными из стали или бетона. Одновременно могут использоваться две такие оболочки. Можно принять меры для ограничения или снижения объема оболочки (или давления в ней) путем концентрации паров, сброса давления, обрызгивания и т. д. Кроме того, фильтрующие и разбрызгивающие системы различных типов могут быть использованы для снижения концентраций радиоактивных материалов, покидающих оболочку. Для дальнейшего сокращения утечек из оболочки могут быть использованы различные методы, в том числе обратная перекачка из вторичной оболочки. Система удержания должна быть защищена от баллистических снарядов. С целью проверки работы системы должны быть проведены ее испытания, которые в дальнейшем должны периодически повторяться.

Обсуждается применение перечисленных выше средств для большинства типов оболочек.

A/276 Estados Unidos de América

Confinamiento y seguridad de las instalaciones de energía nuclear

por W. K. Davis y W. B. Cottrell

Esta memoria examina las medidas de seguridad utilizadas en reactores de producción de energía en los Estados Unidos, tanto en operación como en proyecto. Cuando ha sido posible, se presenta la experiencia adquirida en la operación de algunos dispositivos de seguridad. Además, se discuten las nuevas ideas propuestas para satisfacer exigencias de localización más estrictas. La memoria no estudia los reactores de investigación, experimentales o de propulsión.

En la mayoría de los diseños de reactores de potencia de los Estados Unidos, el combustible se encuentra en forma sólida, y es por lo general un material cerámico, de alto punto de fusión y gran densidad, con un revestimiento muy sólido que resiste la corrosión del agente de refrigeración y retiene todos los productos de fisión. El sistema de refrigeración es generalmente un sistema cerrado, o que puede cerrarse para proveer un sistema cerrado de refrigeración de alta integridad estructural. El reactor y las partes del sistema primario de refrigeración potencialmente radiactivas se cubren con un revestimiento de hormigón. Además, siempre hay alguna forma de confinamiento para cualquier material que pueda escapar del sistema primario de refrigeración y de su revestimiento protector.

Los productos de fisión solamente pueden escapar a esta forma de confinamiento como resultado de algún suceso que provoque el desprendimiento de los productos de fisión del combustible (alta temperatura) y la ruptura del revestimiento y sistema primario de refrigeración. Además, hay posibilidades de accidentes relacionados con la manipulación del combustible. Estos accidentes se estudian separadamente. Dos tipos de fenómenos se examinan generalmente: *a)* un cambio brusco de la potencia de suficiente magnitud para romper el revestimiento y el sistema primario de refrigeración, y *b)* ruptura del sistema primario de refrigeración, y a consecuencia de esto, pérdida del refrigerante y fusión del núcleo del reactor. Estos dos tipos de accidentes se consideran casi imposibles de ocurrir. La memoria presenta consideraciones acerca de estos tipos de accidentes, así como sobre los accidentes en el manejo del combustible de varios

reactores y presenta un resumen de las características esenciales (coeficientes de temperatura, efectos Doppler, etc.) y de las características de diseño (sistemas de control, sistemas de refrigeración de emergencia, etc.) utilizadas en estos reactores.

Los sistemas utilizados para reducir las consecuencias del escape de productos de fisión, aunque es muy poco probable que esto ocurra, incluyen el confinamiento de estos productos en varios tipos de cavidades cerradas, en estructuras de acero u hormigón o en construcciones con un bajo coeficiente de escape. Pueden usarse dos de estos tipos de confinamiento simultáneamente. Se pueden tomar precauciones para reducir o limitar el volumen contenido (o la presión) por medio de supresión del vapor, aligeramiento de la presión, pulverizadores, etc. Además de esto, varios tipos de filtración y sistemas pulverizadores pueden ser utilizados para reducir la concentración de los materiales radiactivos que puedan escapar del confinamiento. Pueden también usarse varios métodos para reducir aún más el escape, incluso el uso de dos cavidades de confinamiento para enviar de nuevo a la cavidad interior cualquier material que haya podido escapar de ella, llegando hasta la cavidad más exterior. El sistema de confinamiento debe también estar protegido contra los proyectiles y debe ser probado (y vuelto a probar periódicamente) para asegurarse de que no existen escapes. Se presentan y discuten estas características en las clases de confinamiento más importantes.

Правила обеспечения ядерной безопасности энергетических реакторов СССР

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ВВЕДЕНИЕ

В настоящее время в СССР работают исследовательские и энергетические реакторы различных типов. Проектируемые и строящиеся в нашей стране реакторы также весьма разнообразны как по своим классификационным особенностям, так и по конкретным инженерным решениям. Со временем количество типов используемых реакторных систем будет, по-видимому, увеличиваться. В связи с этим весьма актуальной оказалась разработка ряда общих требований, выполнение которых при проектировании и выборе эксплуатационных режимов реакторов должно в максимальной степени гарантировать безопасность их работы. С другой стороны, отступление в исключительных случаях от этих общих требований должно приводить к принятию каких-то специальных мер по обеспечению безопасности, запрещению некоторых режимов работы и т. д. Одним из наиболее серьезных вопросов, возникающих при проектировании ядерных реакторов, является вопрос об обеспечении ядерной безопасности, то есть о выработке круга требований и мероприятий, носящих обязательный характер, которые давали бы уверенность в отсутствии быстрых и неконтролируемых разгонов реакторов. Для разработки таких общих положений была создана рабочая группа в составе авторов настоящего доклада. Она подготовила проект «Правил обеспечения ядерной безопасности энергетических реакторов СССР».

В докладе приводятся Правила и их краткое обсуждение.

ПРАВИЛА ОБЕСПЕЧЕНИЯ ЯДЕРНОЙ БЕЗОПАСНОСТИ ЭНЕРГЕТИЧЕСКИХ РЕАКТОРОВ СССР

1. Под обеспечением ядерной безопасности подразумеваются мероприятия, предотвращающие быстрые или неконтролируемые разгоны реакторов.

2. Данные правила не распространяются на критические сборки и реакторы импульсного действия.

3. Рассмотрение аварийных ситуаций должно учитывать возможность наложения во времени двух аварий, одна из которых требует длительного (более часа) времени для ее ликвидации или обнаружения (например, расхолаживание реактора).

4. Никакая авария, включая расплавление твердотопливной активной зоны из-за неисправности системы теплоотвода, не должна приводить к неконтролируемому разгону реактора. Конфигурация, состав и масса делящегося вещества при расплавлении активной зоны должны обеспечивать двойной запас до критического состояния.

5. Изменение ориентации в пространстве, в том числе опрокидывание, транспортных и транспортабельных (при транспортировке) реакторов, а также потопление корабля не должны служить препятствием к срабатыванию органов гашения цепной реакции.

6. Эффективность органов аварийной защиты реактора (АЗ) должна выбираться из следующих соображений:

а) если имеется только один орган компенсации изменений реактивности, то органы АЗ, за вычетом одного из них, и половина автоматического регулятора (АР) должны перекрывать весь диапазон изменения реактивности от рабочей точки до точки, соответствующей максимальной реактивности на кривой ее температурного изменения;

б) если имеется более одного органа компенсации реактивности срабатывающих за время, меньшее времени аварийного расхолаживания, то органы аварийной защиты, за вычетом одного из них, должны перекрывать изменение реактивности между рабочей точкой и температурой теплоносителя на входе в реактор, которая достигается к моменту, когда дальнейшая компенсация реактивности может эффективно обеспечиваться другими органами компенсации

реактивности. При этом весь запас реактивности для полной остановки реактора должен обеспечиваться органами гашения цепной реакции за вычетом любого из них;

в) должно быть не менее двух органов АЗ.

7. В системе АЗ необходимо предусматривать сигнал защиты по периоду нарастания мощности, хотя бы в период пуска реактора и выхода на мощность. Диапазоны действия АЗ по мощности и по периоду должны перекрываться. В цепях АЗ по мощности и периоду должно быть не менее двух измерительных каналов контроля.

8. Запрещается пуск реактора без вывода не менее двух органов АЗ, эффективность которых в сумме должна быть не менее β, причем эффективность системы АЗ без одного из них должна быть не менее $\beta/2$. Это требование должно выполняться при любых рабочих перемещениях системы компенсации реактивности после пуска реактора, в том числе при образовании местных котлов и ксеноновых волн. Подкритичность реактора после взвода указанных органов АЗ и половины стержня АР должна быть не менее 0,5% от k_∞.

9. Срабатывание органов АЗ не должно зависеть от наличия внешних источников питания. При использовании двух каналов срабатывание АЗ должно происходить по сигналу из любого канала. Всегда должно быть не менее двух исправных каналов АЗ.

10. Система АЗ должна допускать проверку исправности всех цепей прохождения аварийных сигналов — от датчиков до приводов исполнительных органов — в процессе работы на мощности без гашения реакции. В цепях датчиков должны быть предусмотрены индикаторы тока или разбаланса токов.

11. Эффективность стержня АР должна удовлетворять одному из двух перечисленных ниже требований:

а) $\rho_{АР} < \beta$;

б) при $\rho_{АР} > \beta$ скорость стержня должна быть такой, чтобы реактивность, равная β, вводилась равномерно за время не меньше 20 *сек*. Одновременно должно выполняться условие: $\rho_{АР} < \rho_{АЗ}$ органов АЗ за вычетом одного из них.

Эти условия не обязательны в тех случаях, когда имеется безынерционный (по сравнению со временем движения стержня) эффект, перекрывающий действие АР.

12. Для разгонов реактора с периодами, соответствующими отработке АР (секундные периоды), производная $\frac{d\rho}{dt}$ (t — температура) должна быть отрицательной в достаточном интервале выше и ниже рабочей точки.

При положительном температурном коэффициенте в какой-либо области температур максимальная скорость ввода реактивности при пуске или разогреве должна быть менее скорости уменьшения реактивности при движении АР. Этим ограничивается мощность при разогреве. Одновременно органы АЗ, за вычетом одного из них, должны перекрывать весь положительный температурный эффект изменения реактивности. В тех случаях, когда органы компенсации реактивности в аварийных ситуациях (за время разогрева) вводятся в активную зону, они могут учитываться в компенсации положительного температурного эффекта.

13. Никакая неисправность в цепях и приводах системы изменения реактивности не должна приводить к скорости увеличения реактивности сверх указанной в пункте 11*б*. Кроме того, должны быть предусмотрены средства, позволяющие при любых неисправностях в цепях системы изменения реактивности реактора обеспечить прекращение ввода реактивности.

14. В проектных материалах и в инструкциях по эксплуатации действующих реакторов обязательно должны иметься разделы, обеспечивающие соблюдение ядерной безопасности при выполнении ремонтных и перегрузочных работ.

15. Для сопоставимости результатов, относящихся к различным реакторам, эффективность органов регулирования и аварийной защиты, а также эффективность всех физических процессов, происходящих в реакторе, должна исчисляться в реактивности ρ. Определением реактивности является формула

$$\nu(1-\rho) = \nu_{кр}. \qquad (1)$$

По формуле (1) должны определяться полный запас реактивности реактора ρ и все отдельные эффекты ρ_i. Составление баланса реактивностей должно производиться по формуле

$$\prod_i (1-\rho_i) = 1-\rho. \qquad (2)$$

16. Все отклонения от настоящих Правил должны быть в каждом конкретном случае специально обоснованы научным руководителем и главным конструктором реакторной установки и утверждены Государственным комитетом по использованию атомной энергии СССР.

Примечание: под β понимается доля запаздывающих нейтронов для данной активной зоны.

ОБСУЖДЕНИЕ

Все правила можно разбить по своему содержанию на три неравных части. Наибольшее число пунктов относится к системе управления реактивностью реактора и особенно к системе АЗ (пункты 4—13). Кроме того, имеются требования, налагаемые на собственно реакторную систему (пункты 3, 4), и, наконец, пункты юридического и административного характера (пункты 1, 2, 14, 15, 16). Обсудим каждую из групп правил.

Прежде чем перейти к обсуждению вопросов, связанных с системой управления реактивностью, отметим пункт 3, который регламентирует количество аварий в системе реактора. Предполагается, что могут быть одновременно две ава-

рии, одна из которых имеет длительный характер, например разгерметизация контура реактора или повреждения механизма, имеющего существенное значение для реактора, и одна авария, носящая кратковременный характер, например отказ в одной из систем управления или кратковременное обесточивание каких-либо механизмов. Предполагается, что более сложные и тяжелые ситуации с наложением аварии, чем указанные в пункте 3, практически не бывают и поэтому при проектировании реакторов не должны рассматриваться. Это общее соображение нашло свое отражение в пунктах, связанных с системой гашения реактивности реактора, так как в них учитывается, что один из стержней-поглотителей либо группа стержней-поглотителей, имеющих общий привод, могут не сработать по прямому назначению.

От системы гашения цепной реакции требуется весьма высокая степень надежности. Поэтому пункты с 4 по 13 ставят ряд очень жестких требований, не требующих специального пояснения. Необходимо отметить только, что во второй части пункта 13 предполагается наличие средств, которые дали бы возможность лицу, занимающемуся управлением установкой, прекратить увеличение реактивности при любых возможных неисправностях в системах. Аналогичным пунктом, но касающимся только системы АЗ, является пункт 9, который предполагает дублирование управления.

Пункты 6 и 8 определяют необходимую эффективность АЗ. При этом предполагается, что после срабатывания АЗ в активной зоне весьма быстро установится температура теплоносителя, близкая к ее значению на входе в активную зону при работе на мощности. Поэтому для компенсации дополнительной реактивности необходимы весьма быстродействующие органы, какими и являются органы АЗ. В дальнейшем изменение температуры в активной зоне и соответствующее изменение реактивности идут уже значительно медленнее и они могут быть компенсированы органами гашения реактивности с существенно меньшим быстродействием. Кроме того, пункт 8 предполагает, что в любой момент работы реактора должна иметься возможность быстро уменьшить реактивность системы на величину порядка β в случае непредвиденного увеличения реактивности при пуске или работе. Обращается внимание на тот факт, что в результате неудачного расположения поглотителей в активной зоне или ксенонных колебаний эффективность АЗ может существенно уменьшиться. Кроме того, выдвигается требование о запасе реактивности до достижения критического положения после подъема АЗ для уменьшения опасности случайной неправильной предварительной оценки реактивности системы перед пуском реактора.

Пункты 7 и 10 выдвигают дополнительные требования к системе АЗ. Требования пункта 10 связаны с тем, что при использовании отдельных приводов для системы АЗ их работоспособность длительное время не проверяется (подвижкой или по прямому назначению). Одновременно все управляющие цепи АЗ также не имеют каких-либо объективных признаков, указывающих на их работоспособность в период стационарной работы реактора. Поэтому регламентируются мероприятия, которые дают возможность хотя бы частично устранить эту неопределенность — обеспечить проверку управляющих цепей АЗ при работающем реакторе.

Пункт 11 дает возможность выбрать эффективность АР таким образом, чтобы исключить быстрые разгоны реактора при любом перемещении стержней АР. При равномерном увеличении реактивности реактора на β за 20 *сек* мощность реактора удваивается через 9 *сек*. Такого времени достаточно для вмешательства лица, управляющего реактором, и срабатывания АЗ.

Пункт 12 выдвигает требование о необходимости саморегулирования активной зоны в области ее рабочих температур, а также регламентирует порядок запуска системы и вывода ее на нормальный режим в тех случаях, когда саморегулирование отсутствует где-либо вне рабочей области температур активной зоны.

Весьма существенными для проектирования ядерных реакторов являются требования пункта 3, по которым конструктивными мерами должен обеспечиваться двойной запас до критической массы при расплавлении активной зоны. В некоторых случаях это накладывает весьма существенные ограничения на конструирование нижнего отражателя активной зоны и нижележащих полостей.

Мы не будем подробно останавливаться на пунктах юридического и административного характера, которые определяют область действия Правил, требования к документации и т. д. Отметим только, что единственной целью пункта 14 является введение единообразия в схему оценки запасов реактивности различных систем.

ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/380 USSR

Nuclear safety regulations for power reactors in the USSR

By G. A. Gladkov *et al.*

Principles for ensuring the nuclear safety of power reactors, which are binding on all research and design organizations, have recently been drafted in the USSR. The principles constitute a code of requirements relating to reactors and their auxiliary systems, and are designed to preclude the possibility of an uncontrolled power increase. The main emphasis is on automatic control and emergency shut-down systems.

The code specifies the measures required to guarantee the impossibility of a fast build-up of reactivity by comparison with the proportion of delayed neutrons during the course of all transient processes, such as: varying the power level, heating-up and cooling (including cooling after emergency shut-down). It also lays down requirements designed to increase the reliability of the emergency shut-down system. The conditions which the construction of the core must satisfy from the point of nuclear safety are also stipulated.

The paper discusses the rules currently in force.

A/380 URSS

Règlements relatifs à la sûreté nucléaire des réacteurs de puissance en URSS

par G. A. Gladkov *et al.*

On a établi en URSS des règlements concernant la sûreté nucléaire des réacteurs de puissance qui sont obligatoires pour tous les organismes de recherche et de construction. Ces règlements fixent un ensemble d'exigences auxquelles doivent satisfaire les réacteurs et les équipements auxiliaires, et dont le but est d'exclure la possibilité d'un emballement du réacteur. Une attention particulière est consacrée aux systèmes de régulation automatique et d'arrêt d'urgence.

Les règlements fixent les normes destinées à garantir l'impossibilité d'une introduction rapide de réactivité comparable à la part des neutrons retardés dans tous les processus non stationnaires: variation de puissance, échauffement et refroidissement (entre autres refroidissement après arrêt d'urgence). On indique les exigences qui permettent d'augmenter la sûreté du système d'arrêt d'urgence. On expose les conditions auxquelles doit satisfaire la construction de la zone active du point de vue de la sûreté nucléaire.

Les règlements en vigueur sont discutés dans le mémoire.

A/380 URSS

Normas de seguridad nuclear de los reactores de potencia de la URSS

por G. A. Gladkov *et al.*

En la URSS existen actualmente normas de seguridad nuclear de los reactores de potencia que son obligatorias para todos los organismos de investigación y de proyecto. Dichas normas constituyen un código de los requisitos que deben satisfacer los reactores y sus sistemas auxiliares, y su objeto es evitar la posibilidad de un aumento incontrolado de la potencia. Se presta principal atención a los sistemas de control automático y de parada de emergencia.

Esa reglamentación especifica las medidas que garantizan la imposibilidad de una introducción rápida de reactividad, en comparación con la fracción de neutrones retardados, para todos los procesos transitorios, como: variación de la potencia, calentamiento y enfriamiento (incluido el enfriamiento subsiguiente a la parada de emergencia). Se subrayan los requisitos que permiten aumentar la confiabilidad del sistema de parada de emergencia. Se determinan las condiciones que, desde el punto de vista de la seguridad nuclear, debe satisfacer el diseño del núcleo.

La memoria contiene algunas consideraciones sobre las normas en vigor.

Recent developments in the application of meteorology to reactor safety

By D. H. Pack, J. K. Angell, I. Van der Hoven and D. H. Slade *

The relation of meteorology to the nuclear industry has developed in two rather different directions. The increasing number of "conventional" (e. g., pressurized or boiling water) reactors whose nuclear and effluent characteristics are rather well understood has created a need for general estimates of the local dilution climate expected in a wide variety of locations. This information should, if possible, be developed in terms of probabilities of various broad dilution regimes, with particular emphasis on the slower dilution situations. Increased experience and the development of better models of reactor core behavior and fission product release amounts and characteristics have reduced the need for the more detailed meteorological studies at all reactor sites.

On the other hand the development of new reactor types, particularly for aerospace applications and the increased program of reactor testing has emphasized a variety of problems in meteorology including diffusion over much greater distances, say to 100 km, and the concomitant trajectories of such effluents. There is also increased interest in the very short range diffusion, say to 100 m, and the effect of terrain or building structures in changing the diffusive character of the atmosphere.

Thus, present meteorological programs are now more closely designed to match the specific problems of a particular reactor installation. Since in many instances the acquisition of on-site meteorological data is limited, there is considerable need to utilize existing meteorological records, in combination with a minimum site program to best estimate the local diffusion conditions.

In this review we will stress several techniques for determining, rapidly and with minimum costs, the required diffusion data. In addition, much of the paper will be devoted to a discussion of a relatively new technique, most applicable to large research or operational installations, for determining air trajectories and diffusion over distances of 20 to 100 km.

DIFFUSION CLIMATOLOGY

Considerable attention is currently being given to the regional and climatological description of atmospheric measurements related to the dilution capacity of the air within the first few thousand feet of the surface. Hosler's [1] study of low level inversion frequency in the contiguous United States was an attempt to quantify, on a geographical basis, the inhibiting effect of low level night-time radiation inversions upon the dilution capacity of the atmosphere. Figure 1 shows the annual surface inversion frequency in per cent of total hours for the United States. Of particular note are the inversion maxima in the south-eastern and western portions of the country, with lower frequencies in general along the coast lines. Holzworth's [2] study on mean maximum mixing depths is a companion effort to index the daytime phenomenon of enhanced atmospheric dilution near the surface due to insolation effects. The determination of the daily mixing depth involves the daily maximum temperature as obtained from standard hourly weather observations and the depth of the preceding night-time surface inversion layer as obtained from routine radiosonde data. From a temperature-height diagram, as shown in Fig. 2, the height at the intersection of the typical daytime temperature lapse rate (lower solid line) and the preceding night-time surface inversion (dashed line) determines the maximum mixing depth for the day. Figure 3 shows the mean maximum mixing depth during the fall season for the United States. Interestingly, the figure shows that the western portion of the country has an unlimited mean maximum mixing depth, i.e., good

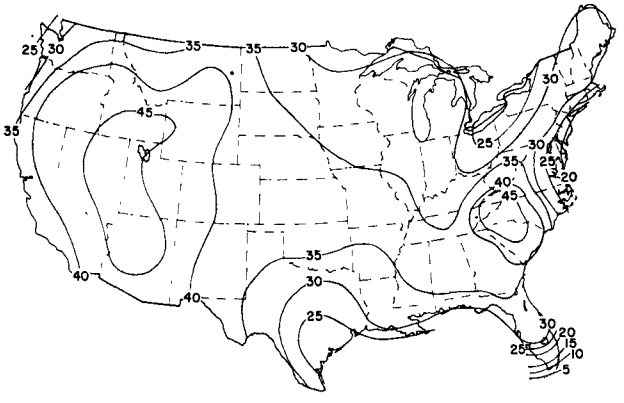


Figure 1. Annual surface inversion frequency in per cent of total hours for the United States

* US Weather Bureau, Washington, D.C.

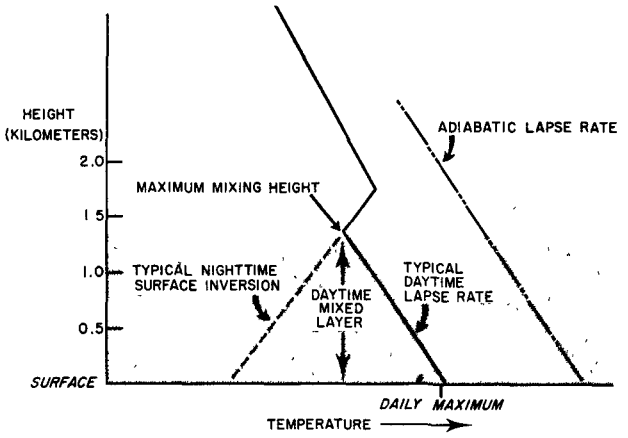


Figure 2. Maximum mixing depth determination

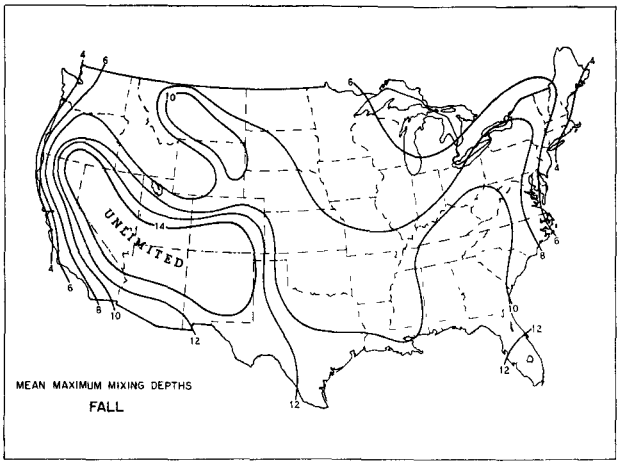


Figure 3. Mean maximum mixing depths for fall for the United States

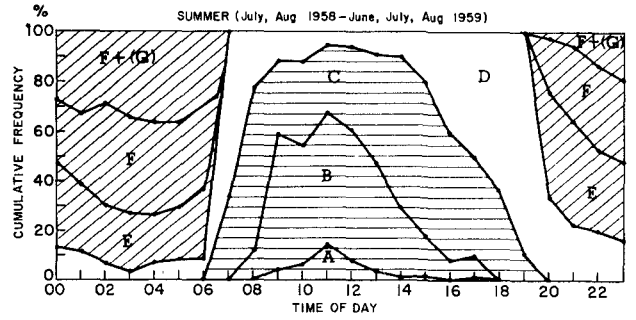


Figure 4. Diffusion index for summer for Cape Kennedy, Florida

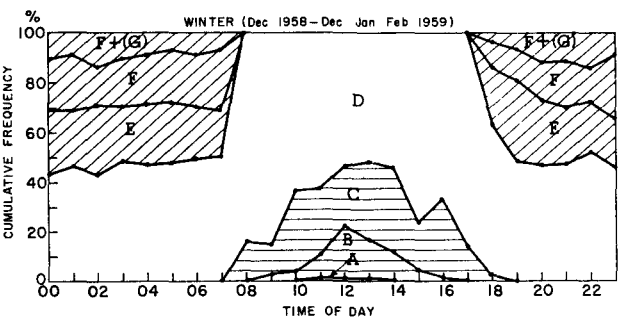


Figure 5. Diffusion index for winter for Cape Kennedy, Florida

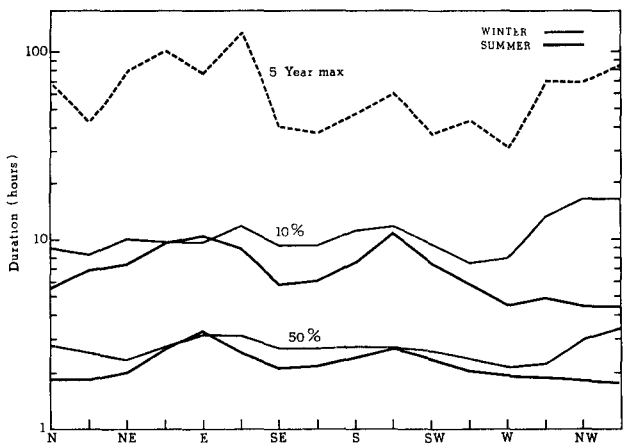


Figure 6. Wind direction persistence for Daytona Beach, Florida (67½° sector)

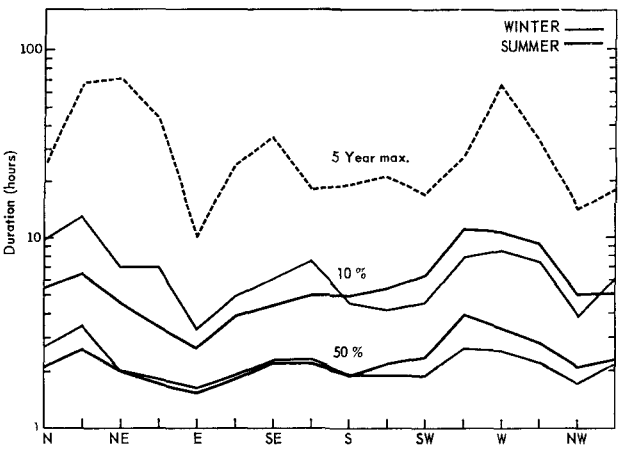


Figure 7. Wind direction persistence for Point Mugu, California (67½° sector)

diffusion during the daytime, whereas the previous figure shows a high frequency of poor diffusion during the night-time for the same general area. This broad-scale type of assessment of dilution capacity provides useful information for the siting and engineering of nuclear facilities especially if any routine radioactive effluent is anticipated.

The use of conventional meteorological observations to infer diffusion regimes was suggested by Meade [3] and Pasquill [4] based on wind speed and cloud height and amount, and a further adaptation by Turner [5] using sun elevation angle. Thus for example, by the computer processing of 18 months of standard hourly weather observations from Cape Kennedy, Florida, analyses such as shown in Figs. 4 and 5 are obtained. Plotted in the figures are the comulative frequencies of category A (very good diffusion) to category G (very poor diffusion) as a function of time of day. Of interest

is the rapid transition from poor to good diffusion at sunrise during the summer season while during the winter large frequencies of neutral stability (category D) occur with a subsequent decrease in very poor and very good diffusion. Gifford [6] has elaborated on the technique by relating the diffusion categories to the standard deviation of the cloud dimensions as a function of distance which in turn can be used to compute down-wind concentrations.

In connection with the emission of radioactive effluents over time periods ranging from hours to days, an increasing amount of interest has been expressed in quantitative wind direction persistence measurements. Using standard hourly wind direction reports, the probability that a wind direction will persist within a sector of 22 ½°, 67 ½°, and 112 ½° (1, 3, and 5 points of a 16 point compass) centered on a given wind direction for a given number of hours was computed by machine methods. Figure 6 shows the wind persistence for the 67 ½° sector for Daytona Beach, Florida, for winter and summer. It is to be noted that at the 10% probability level the winds tend to persist for about 10 hours and are more persistent from the northwesterly directions in winter. The longest single event of persistence in this sector for five years of data was a period in summer of 125 h centered on ESE. The overall persistence values for Daytona Beach may be compared with those of Point Mugu, California, in Fig. 7 where persistence is somewhat less, since the 10% line is considerably below the 10 h value. A definite peak persistence in the seaward and landward directions of this coastal station is indicated, however. In order to identify regional and seasonal similarities and anomalies with regard to wind direction persistence and to devise a wind persistence index, five years of routine hourly wind data from 64 stations in the continental United States have been processed in this manner and are now being further analyzed in order to present a wind persistence climatology.

DIFFUSION ESTIMATES FROM WIND DIRECTION FLUCTUATION DATA

The basic problem in the field of applied diffusion studies has been the direct estimation of diffusion parameters from simply obtained meteorological data. In this discussion, we will consider only low-level, continuous effluent releases of a few hours or less in length. Much of the diffusion theory and most of the diffusion experiments accomplished to date have dealt with such configurations.

The door to today's simple, practical diffusion estimation methods was opened in 1921 by G. I. Taylor's [7] now classic paper "Diffusion by Continuous Movements". The principles of this statistical treatment of the Lagrangian motions of dispersing particles were used by Sutton [8] to construct his now famous series of semi-empirical diffusion prediction equations. The meteorological parameters required for diffusion estimation from these equations are wind component fluctuation and wind speed gradient data. Considering the extent to which these equations were used in field evaluations, surprisingly little was done to evaluate the terms directly. This was particularly true of the wind fluctuation terms, the determination of which was quite time consuming. As a result, the few experimental determinations of Sutton's diffusion coefficients were used over and over again, many times in an uncritical way, by those involved in reactor safety analyses.

In 1932, Giblett [9] and later Smith [10], of the Brookhaven National Laboratories, presented simple, if subjective, systems for categorizing wind direction trace types into a few basic types which had direct implications to diffusion estimation. Smith assumed that large wind direction fluctuations would be associated with wide plumes and small fluctuations with rather narrow plumes. These assumptions fit quite well within the framework of Taylor's approach, which considers diffusion as proportional to the turbulence level. These trace types determined on the basis of the wind direction trace width and, to some extent, according to the appearance of the trace, have been used to classify diffusion at the Brookhaven Laboratories ever since their conception.

In 1953, Holland [11] sought a climatological evaluation of the Sutton parameters directly from measured meteorological data. In order to measure the wind direction fluctuations in a simple but rational way, Holland used the total angle through which the wind moved in a given time period. He showed that this "range" value could be considered as an estimate of the wind direction standard deviation, a quantity which he could use in the estimation of C_y and C_z. He still, however, included the measures of wind speed gradient called for by Sutton.

During the summer of 1956, an intensive program of turbulence and diffusion measurement was carried out over a very flat site in the central United States. Project "Prairie Grass", as these experiments were called, furnished the most thorough set of field data for estimating diffusion to distances of 800 m that had been assembled at that time. Abandoning the traditional approach in seeking diffusion parameters, that involving the measurement of wind speed and vertical temperature gradients, Cramer [12] analyzed all of the approximately 70 Prairie Grass diffusion experiments directly in terms of the standard deviation of the horizontal wind direction fluctuations. While his approach consisted of completely empirical curve fitting procedures, he demonstrated rather conclusively that wind direction fluctuations contained sufficient information for estimating diffusion under a wide range of meteorological conditions. Cramer maintained that

"... the distribution of azimuth wind direction appears to contain implicit information on site roughness and other factors affecting dispersal and shows promise of serving as a universal diffusion index ...".

Simultaneously with the publication of Cramer's work, Hay and Pasquill [13] presented the results of experiments designed to show directly the relationship between wind direction fluctuations and particle dispersion in the vertical. They found not only that the downstream material distribution was Gaussian, which was generally expected, but that the distribution was predictable from the distribution of vertical wind directions over the time of release. This suggested that the Lagrangian motion data, called for by Taylor, could be estimated from more easily obtained Eulerian wind measurements. This was, of course, a deduction which did not disagree with Cramer's findings.

Carrying their work further, Hay and Pasquill [14] showed how simplifying assumptions relating to the shape and time scales of Eulerian and Lagrangian autocorrelation could be used objectively to allow the direct interpretation of Eulerian wind data in terms of plume spread. The result of this work was to give additional theoretical justification to the more empirical methods of direct usage of wind direction data for diffusion estimation.

In the last few years, three comprehensive series of diffusion experiments have been carried out in the United States. During the period 1959–1962, a large number of diffusion field tests, collectively referred to as the Green Glow-30 series, were conducted at the Hanford Atomic Products Operation at Richland, Washington (Barad and Fuquay [15]; Fuquay, Simpson and Hines [16]), 46 of which were subjected to intensive analysis. The second series of tests, conducted in 1961, were carried out at the National Reactor Testing Station in Idaho Falls, Idaho (Islitzer and Dumbauld [17]). The third series was carried out at two locations, Cape Kennedy, Florida and at the Vandenberg Air Force Base in California (Haugen and Taylor [18]).

The unique feature of each of these three experiments was the inclusion of horizontal wind direction fluctuation measurements, in the form of the standard deviation of the wind direction over the period of the release, as one of the primary meteorological estimators of diffusion. Thorough studies of these experiments indicate that a generally quite adequate diffusion climatology may be prepared for a nuclear site using no other data than that available from a common wind sensing and recording system. The use of the phrase "quite adequate" requires some discussion in the context of the site diffusion climatology. Preliminary estimates indicate that greater than 70% of the observed downstream concentration values in the various experiments can be predicted within a factor of ± 2 from wind considerations alone. Virtually all of the predicted values fall within a range of ± 5 of that observed. It is rare indeed that source strengths can be estimated during the initial phases of site and nuclear system evaluation with greater accuracy than possible for diffusion coefficients.

Within the next year, a number of simple techniques for nuclear site evaluation using simple wind equipment should be forthcoming. We, ourselves, are preparing one. The common characteristics of these techniques, beside their direct use of wind fluctuation data, will be their simplicity and low cost in terms of equipment and data handling. It should soon be possible, for the first time, to construct rational climatologies of diffusion for most sites, climatologies which, moreover, will be directly comparable between stations. The computation of wind direction standard deviation, quite recently an extremely time consuming process, may now be accomplished by rather simple electronic methods. Jones and Pasquill [19] and Culkowski and Turner [20] describe two different approaches to this problem. In lieu of even this sophistication, the wind direction range may be used to estimate the standard deviation when climatological estimates are desired (Markee [21]).

SHORT RANGE DIFFUSION

Determination of atmospheric dilution over distances less than 100 m in other than flat terrain is extremely difficult. If the emission takes place in a building complex or where abrupt terrain changes occur, the aerodynamic characteristics of the flow may be much more important than the random turbulent diffusion. One successful approach to a quantitative analysis of this problem has been the use of wind tunnel modelling of the concentration distribution around both prismatic building shapes and a typical reactor containment shell (Halitsky [22, 23]). This work involved release of a gaseous tracer (SO_2) from vents or stacks in a model building and measuring the three-dimensional concentration field around these models. Of particular interest was the development of a diffusion coefficient scaled to model dimensions such that application to full scale structures is feasible. Of the measured results in the wind tunnel, perhaps the most interesting is the relatively high concentrations up-wind of the point of emission when this emission is near ground level and the air flow is disturbed by nearby buildings (Fig. 8).

In many instances, however, it may not be feasible to carry out wind tunnel tests or less elaborate information will suffice. In such cases smoke may be used as a visual tracer. Recent work by Culkowski [24] has shown how conventional film and cameras can be used with appropriate filters to obtain long time exposures of emissions. This method permits a vivid

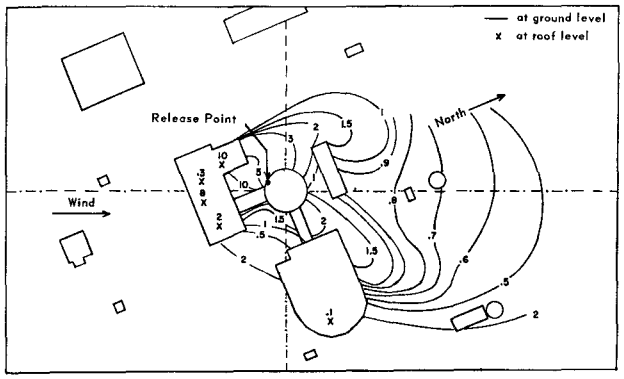


Figure 8. Dilution isopleths within a building complex—Wind tunnel tests

Isopleths are relative values. Note high concentrations upwind for this release on the upwind side at the base of the reactor shell

determination of persistent anomalies created by terrain and building effects, free from the random instantaneous fluctuations.

SMOKE TRACERS FOR MEDIUM RANGES

In addition to using this technique for short range diffusion where conventional tracer techniques are either difficult or too costly, it can be used to determine longer range diffusion coefficients. Gifford [25] has shown how time exposure photography of smoke plumes can be used, quantitatively, to obtain diffusion coefficients. If the smoke source is of sufficient strength to be visible, the method can be used over distances of several km. In the absence of time-averaged photographs, overlays of a series of instantaneous plume "snapshots" have been used to simulate this time averaging with excellent results (Bowne [26]). The few experiments mentioned here indicate the potential of this technique which has, surprisingly considering its simplicity and low cost, been exploited very little.

THE USE OF BALLOON-BORNE RADIO BEACONS TO OBTAIN AIR TRAJECTORIES AND DIFFUSION DATA ON THE MESOSCALE

A basic problem with regard to nuclear sites is the estimation of the direction of travel of an effluent, whether released accidentally or intentionally, and the degree to which this effluent will be diffused by the atmosphere. In accord with the general development of the atmospheric sciences, such determinations have, in the past, been based upon wind statistics obtained at fixed points (Eulerian network). However, fundamentally, atmospheric diffusion is a function of the wind statistics derived from the movement of individual air parcels (Lagrangian statistics). Furthermore, while Eulerian wind statistics may be suitable for diffusion estimates over small times and distances, for the longer times and distances there are almost insurmountable difficulties inherent in this technique. For example, as the effluent spreads in the vertical, wind direction and speed changes are encountered which cannot be measured by a surface network of anemometers and which are difficult and expensive to determine by existing fixed-point, vertical wind-sounding methods.

In an effort to obtain Lagrangian statistics directly, we have developed a small, constant volume balloon. The balloon (called a tetroon) is constructed of the nearly inelastic material Mylar, and thus tends to conserve its volume regardless of the excess of internal gas pressure with respect to air pressure (superpressure). Inasmuch as there is little variation in balloon volume, there is also little variation in buoyancy force, and the tetroon will tend to float along a surface of constant density with little variation in mean flight altitude. The balloons actually utilized were tetrahedral in shape (hence the name tetroon), 60 in on a side, with a skin thickness of 0.002 in and a nominal volume of 1 m^3. Balloon inflation was carried out through a conventional bicycle valve. Very precise weigh-off is required with these balloons inasmuch as 1 g of ballast can lower the mean flight altitude by 100 ft (Angell and Pack [27]).

Initially the tetroons were positioned by radar through the use of passive reflectors. However, at the usual float altitude of the tetroons (1 000–3 000 ft) the problem of ground clutter is very serious and the desirability of using a positive electromagnetic signal to track and identify the floating tetroon was soon recognized. Consequently, a transponder was developed which could be triggered by the Weather Bureau WSR-57 radar network (Pack [28]). The transponder was 1.3 in wide, 1 in deep, and 10 in long (plus a 13-in antenna rod), and its weight, minus batteries, ranged from 112 g to 156 g depending on the density of the styrofoam weatherproofing. The transponder was powered by a 1 ½ V flashlight battery and a 22 ½ V transistor-radio battery. High energy alkaline batteries were used, with a power output of about 1 W. With the batteries the total pay-load was only about 265 g. The nominal 403 mc/s signal from the transponder could be varied in frequency, thus permitting positive identification when two transponders were aloft at the same time. The 403 mc/s signal was detected by a rotatable V-type yagi antenna, then fed into a high-gain amplifier especially built for this purpose, thence to a FMQ-2 receiver modified for AM reception. This signal was then fed into the radar video circuit where the range was determined by time comparison of the radar and transponder pulses in the usual way. With this tetroon transponder system individual tetroons can be tracked for at least a day and to a distance of more than 100 miles from the radar.

The most extensive use of the tetroon transponder system has been in the Los Angeles Basin (Pack and Angell [29]). Figure 9 shows tetroon trajectories from

Long Beach at heights of 1 000–2 000 ft. To be noted are the large directional shifts along these flights, due mainly to land and sea breeze effects. There would be great difficulty in deriving many of these trajectories from fixed-point wind observations, and thus the tetroon appears as a superior means of tracing the trajectory of an effluent.

In addition to providing an estimate of the direction of travel of an effluent, tetroons may be utilized to yield an estimate of the diffusive capabilities of the atmosphere. The most obvious way to estimate lateral (horizontal, cross stream) diffusion from tetroon flights is to release tetroons in quick succession and then determine the root mean square value of the lateral distance between individual trajectories and mean trajectory at various downstream distances. Table 1 shows lateral standard deviations obtained from series of tetroon flights within the Los Angeles Basin. The series were determined in such a way that a mean wind could reasonably be defined for each series. The number of tetroon flights in each series varied from four in series 2 and 5, to ten in series 6. Within each series the time interval between first and last tetroon release equalled about 8 h in series 5, 24 h in series 1, 2, 3 and 4, and 48 h in series 6. For these flights in the Los Angeles Basin the standard deviations are large, averaging 50–70 per cent of the downstream distance. It is emphasized, however, that we are dealing here not only with a continuous source acting over a period of about 24 h but also with a continuous sample for a similar period. Thus, the lateral standard deviations reflect not only what might be considered pure diffusion, but also the effect of changes in the mean wind direction with time, and the strength of the diurnal wind regime in the Los Angeles Basin is well known. A regression line fitted to these lateral standard deviations indicates that, over downstream distance of 2–50 km, the lateral standard deviation is proportional to downstream distance to the 0.90 power.

It has been shown by Gifford [30], Pasquill [31] and that others lateral diffusion (lateral standard deviation) can be estimated from running-means of the lateral wind fluctuations and that this is, theoretically, fully equivalent to using Taylor's [7] original formulation involving double integration of the autocorrelation coefficient. The relevant formula is

$$\sigma_{y(T)} = \sqrt{\left(\overline{v'^2}_{(T)} T^2\right)} \qquad (1)$$

where $\sigma_{y(T)}$ is the lateral standard deviation after travel time T, and $\overline{v'^2}_{(T)}$ is the variance of the lateral wind fluctuation averaged over the travel time T.

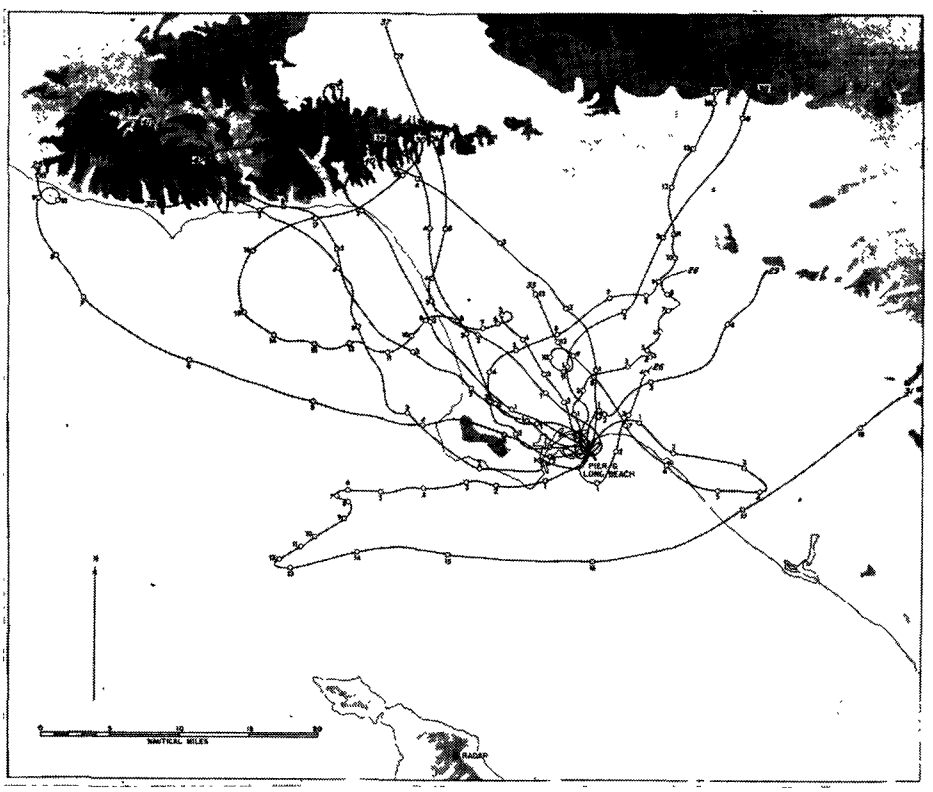


Figure 9. Tetroon trajectories of two or more hours' duration for tetroon releases from Long Beach within the Los Angeles Basin

Tetroon positions indicated at 1-hour intervals. Topography shading (progressively darker) for heights exceeding 500, 1 000 and 2 000 feet

Table 1. Lateral standard deviation (km) as a function of downstream distance for a series of Los Angeles tetroon flights

Los Angeles series	Downstream distance (km) 2	3	5	10	15	20	25	30	35	40	45	50
1			4.2	7.5	11.0	12.7	17.2					
2			4.4	8.1	9.8							
3			3.2	4.7	8.5	13.5	14.4	15.3	16.6			
4			2.6	5.2	7.7	10.0	12.2	14.1	15.4	16.5	19.4	23.9
5			3.4	7.6	10.4	13.8	16.3	19.4	21.0			
6			2.8	4.6	6.2	9.7	12.1	14.0	14.9			
Average .			3.4	6.3	8.9	11.9	14.4	15.7	17.0			

The use of Eq. 1 to obtain diffusion estimates from a single trajectory has been considered by Angell [32]. First of all, it should be noted that a single set of lateral wind fluctuation (v') statistics of infinite length in a stationary, homogeneous turbulence field would, in theory, provide complete information on lateral diffusion. However, for a finite length of v' in a steady, homogeneous turbulence field we should expect the σ_y value to always be less than the true value. In the case of real atmospheric data we should expect differences due not only to the truncated record but also to changes in the turbulence field (e. g., changes in the mean wind), to the statistical manipulation of the data, and to radar positioning errors, which affect the v' statistics more than the trajectories *per se*. On the other hand, the lateral standard deviations derived in this way refer to a sampling time corresponding to the travel time T, which in turn is comparable to the sampling time utilized by previous investigators. Thus, the lateral standard deviations derived from Eq. 1 might be expected to more closely resemble conventionally obtained continuous point-source diffusion data than the lateral standard deviations obtained from serial tetroon releases.

The solid lines in Fig. 10 represent lateral standard deviations (σ_y) as functions of downstream distance (x) as derived from individual tetroon flights through the use of Eq. 1. In general, the tetroon flights were made at heights near 2 000 ft. For clarity in Fig. 10, the solid lines have been drawn as straight lines even though there was some slight tendency for the lines to be concave downward. The lateral standard deviations were computed for travel times as large as 20 per cent of the duration of the v' statistics, but this is probably extreme, and when longer tetroon flights are available, it would be desirable to limit evaluation to 10 per cent of the duration of the v' statistics. An obvious drawback of this method is that it requires very long tetroon trajectories to obtain diffusion data at even moderate distances downstream.

Figure 10 shows that, on the basis of the tetroon flights, the ratio of lateral standard deviation to downstream distance varies from a value of 0.02 for daytime flights over the sea from Wallops Island (W) and a night-time flight over the desert at Yucca Flat, Las Vegas (LV-6), to an average value of 0.5 for day and

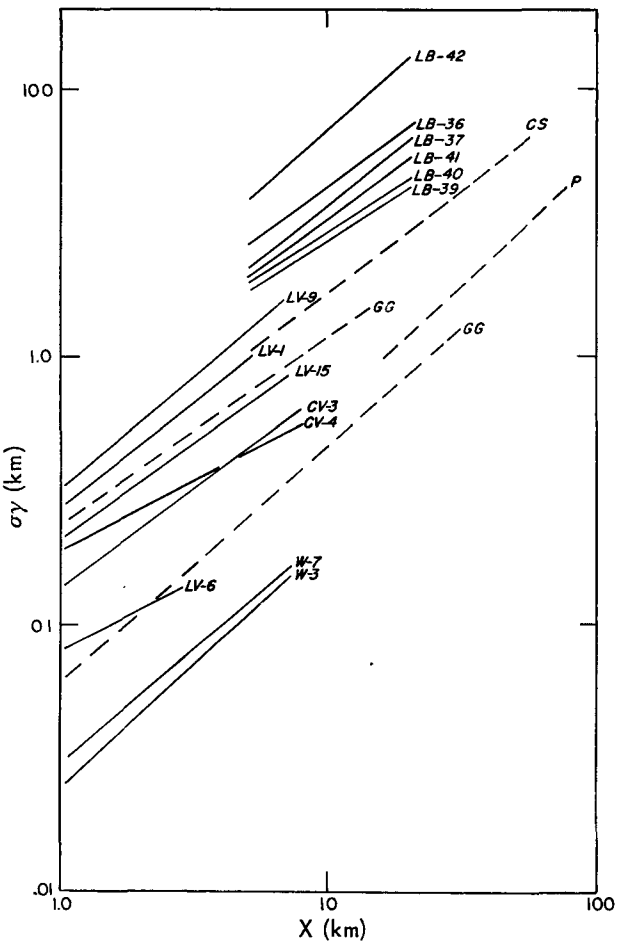


Figure 10. Lateral standard deviation as a function of downstream distance as derived from running means of the lateral velocity components along individual tetroon flights from Long Beach (LB), Las Vegas (LV), Covington, Kentucky (CV), and Wallops Island (W)

The dashed lines represent continuous point source lateral diffusion estimates obtained by Crozier and Seely (CS), Pasquill (P), and obtained from Project Greenglow (GG)

night flights from Long Beach (LB), within the Los Angeles Basin. Long Beach flight 42 undergoes a large direction shift due to land and sea breeze effects and in this case the above ratio approaches one. For comparison, the dashed lines in Fig. 10 indicate the lateral standard deviations obtained by Crozier and Seely [33], Pasquill [34], and obtained from Project Green Glow as reported by Barad and Fuquay [15].

The Crozier and Seely evaluations refer to desert terrain and approximately agree with the lateral diffusion derived from Las Vegas tetroon flights made near noon on different days (LV-1, LV-9). Flight LV-15 was made at 9 a.m. Pasquill's data were obtained in slightly hilly terrain near Porton, England, and are in general agreement with the results derived from Covington, Kentucky, tetroon flights (CV-3, CV-4) in similar terrain (Pack [28]). It might be noted that flights CV-3 and CV-4 were released simultaneously, and consequently the degree of similarity in the lateral standard deviations derived from these two flights is evidence for the representativeness of this technique. The appropriateness of the tetroon technique is also

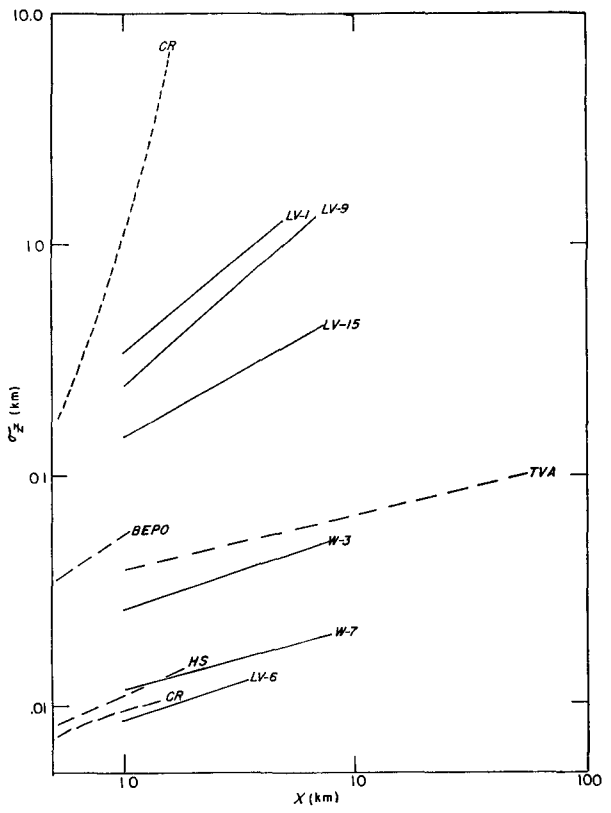


Figure 11. Vertical standard deviation as a function of downstream distance as derived from running means of the vertical velocity components along individual tetroon flights from Las Vegas (LV) and Wallops Island (W).

The dashed lines represent continuous point source vertical diffusion estimates obtained by Cramer (CR), Hilst and Simpson (HS) and from British Experimental Pile and Tennessee Valley Authority

suggested by the similarity in the rate of change of lateral standard deviation with downstream distance derived from the tetroon flights and from conventional means. In most cases the lateral standard deviation obtained from tetroon flights is proportional to downstream distance to the 0.8–0.9 power, although there are striking exceptions (CV-4, LV-6).

The solid lines in Fig. 11 represent the vertical standard deviation (σ_y) as a function of downstream distance (x) as derived from individual tetroon flights through the use of Eq. 1. Because of the difficulties involved in obtaining accurate tetroon vertical velocities from the WSR-57 radars at Long Beach and Covington, these data ave been omitted from Fig. 11. Otherwise, the same tetroon flights are presented in Fig. 11 as in Fig. 10.

Figure 11 shows that, on the basis of the tetroon flights, the ratio of vertical standard deviation to downstream distance varies from a value of about 0.01 for daytime flights over the sea and night-time flights over the desert to a value of 0.1 for daytime flights over the desert near Las Vegas. In the former case the ratio varies considerably since there is a very small increase in vertical standard deviation with downstream distance (vertical standard deviation proportional to downstream distance to the 0.27 power). However, the fact that the flights near noon at Las Vegas yield a power of 0.8–0.9 certainly suggests a variation of this power with atmospheric stability. The dashed lines in Fig. 11 represent direct measurements of vertical standard deviation obtained by Hilst and Simpson [35], Stewart, Gale, and Crooks [36] and by the Tennessee Valley Authority, as reported by Gartrell, Thomas, and Carpenter [37], as well as indirect estimates of standard deviation obtained by Cramer [38] in stable and unstable conditions. The Gartrell *et al* data and data of Hilst and Simpson refer to stable conditions, the Stewart *et al* data to slightly unstable conditions. Once again the appropriateness of the tetroon technique is indicated by the similarity in the rate of change of vertical standard deviation with downstream distance as derived from the tetroon flights and from conventional means. One possible discrepancy involves the observation that the tetroon flights in desert areas under very unstable conditions do not yield nearly as rapid an increase in vertical standard deviation with downstream distance as do the results obtained by Cramer in unstable conditions. We probably see here a bias inherent in the tetroon data in that there is always a force tending to return the tetroon to its equilibrium floating surface and this tendency, which reduces the vertical amplitude of tetroon oscillations with respect to air parcel oscillations, will lead, through Eq. 1, to too small a rate of increase of vertical standard deviation with downstream distance.

A proper evaluation of the possible error involved in utilizing individual tetroon trajectories to estimate

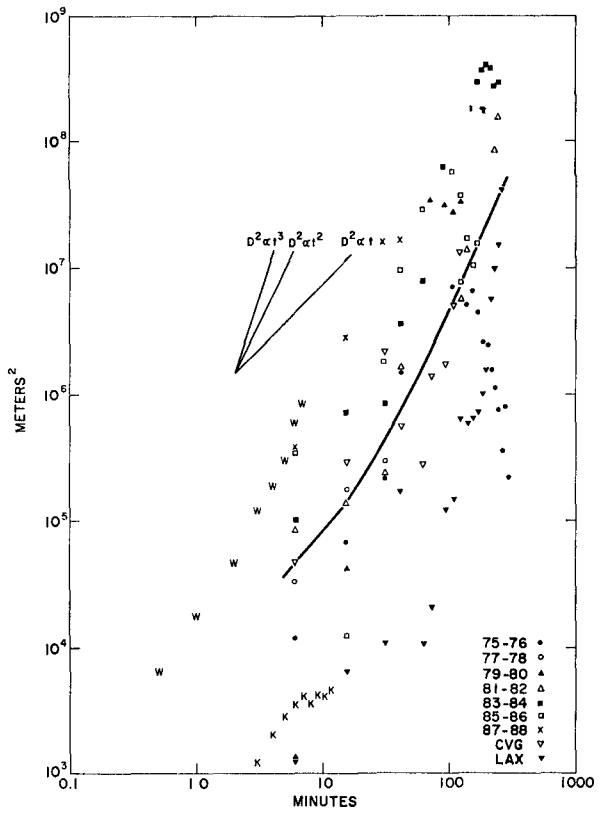


Figure 12. Square of the separation distance between pairs of tetroons as a function of time.

Numbered pairs represent simultaneous tetroon flights within the Los Angeles Basin, CVG indicates flights from Covington, Kentucky, and LAX indicates flights made by Holzworth *et al* over Los Angeles. Results obtained by Wilkins (W) with no-lift pibals near the ground, and by Kellogg (K) with smoke puffs in the stratosphere, are shown for comparison

lateral diffusion (or ground concentration) must await simultaneous tetroon and tracer experiments. The only comparison made so far has been reported on by F. Pooler * of the Weather Bureau Research Station at Cincinnati, Ohio. He found the difference between σ_y measured from a surface release of fluorescent particle at St. Louis, Missouri, and from a tetroon flight made at the same time, to be less than 10 per cent.

The second basic type of diffusion is the so-called relative diffusion, which involves the rate of growth of an individual cluster of particles. Basically, this smoke-puff type of diffusion is a function of the increase in distance between pairs of particles and thus requires, having two trajectories at the same time as a minimum. The use of tetroons to obtain estimates of relative diffusion is in its infancy, and only a few cases are available for presentation here. It is of course, obvious that the tetroons must possess very nearly the same equilibrium floating surface, else the wind shear in the vertical will cause a fictitiously large rate of balloon separation with time. Consequently, for relative diffusion experiments the tetroons must be weighed-off with great care.

Figure 12 shows the square of the distance between pairs of simultaneously released tetroons as a function of time after release of the tetroons. The numbered pairs of flights were made from the Palos Verdes Hills in Los Angeles, while the pair labelled CVG refer to flights at Covington, Kentucky, and the pair labelled LAX to flights made over Los Angeles by Holzworth *et al* [39]. In the latter case tetroon positions were obtained by following the balloons with cars. Included for comparison in Fig. 12 are no-lift pibal data obtained by Wilkins [40] near the ground and smoke-puff data obtained by Kellogg [41] in the stratosphere. The solid line represents the medial value of the square of the tetroon separation distances with respect to time, and is seen to be intermediate to the findings of Wilkins and Kellogg. The short sloping lines in Fig. 12 indicate the power of the time to which the square of the separation distance is proportional. The medial line suggests a power varying from 1.5 at a time of 10 min to a power of 2.5 at times exceeding 100 min. However, on a few of the paired tetroon flights the tetroons actually approached one another at times after release exceeding 100 min.

From the range of tetroon separation distances in Fig. 12, it is apparent that many more simultaneous tetroon releases must be made before a reliable estimate of relative diffusion on the mesoscale can be obtained, or attempts can be made to determine the manner in which relative diffusion varies with atmospheric conditions or underlying terrain. However, with a very careful weigh-off procedure, there appears to be no reason why such information cannot be obtained in the future.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/714 OMM

Progrès réalisés récemment dans les applications de la météorologie à la sûreté des piles

par D. H. Pack *et al.*

Les récents progrès de la technique nucléaire ont nécessité et fourni de nouveaux moyens et de nouvelles méthodes météorologiques qui contribuent à déterminer de façon plus exacte dans quelle mesure l'emplacement d'un réacteur peut se révéler satisfaisant et à évaluer avec plus de précision la diffusion atmosphérique; ces progrès ont également fait apparaître de nouvelles méthodes pour mesurer l'écoulement de l'air.

On a de plus en plus tendance à élaborer des programmes météorologiques pour répondre aux besoins de certains types d'installations nucléaires et résoudre les problèmes particuliers que posent ces installations.

Parmi les nouvelles méthodes d'approche décrites dans le mémoire pour le choix des emplacements de réacteurs, il est fait mention des analyses du gradient vertical effectuées sur le plan national et des méthodes permettant de déterminer la persistance des régimes de vents. L'utilisation d'observations météorologiques classiques pour déterminer les régimes de diffusion est décrite et illustrée pour divers emplacements.

L'application de données concernant les variations du vent au calcul des coefficients de diffusion a été perfectionnée et l'on a mis au point des méthodes empiriques permettant d'utiliser les régimes classiques des vents.

Le document décrit des méthodes fondées sur des photographies de fumées qui permettent d'effectuer des analyses quantitatives, ainsi que des exemples pratiques d'études au traceur se rapportant à des problèmes particuliers.

L'une des plus récentes méthodes météorologiques applicables aux sites de réacteurs consiste dans l'utilisation de radars portatifs emportés par des ballons pour déterminer les trajectoires des particules d'air et obtenir des données sur la diffusion dans un rayon pouvant atteindre 160 km (100 milles). Cette méthode ainsi que les possibilités qu'elle offre sont décrites de façon très circonstanciée.

A/714 ВОЗ

Последние достижения в области использования метеорологии в обеспечении безопасности работы реакторов

Д. Х. Пак *et al.*

Для достижения современных успехов в ядерной технике потребовалось разработать новые метеорологические приборы и методы, способствующие более точной оценке пригодности площадок для размещения реакторов, проведению более точных расчетов атмосферной дисперсии, кроме того, потребовалось создать новые методы измерения воздушных потоков.

Все возрастает тенденция составлять программы метеорологических измерений в соответствии с объемом и особыми задачами конкретных ядерных установок.

Описываются новые способы решения задачи размещения реакторов, основанные на анализе величины вертикального градиента давления в масштабе всей страны и методах определения постоянства режимов ветров. Описывается и иллюстрируется на примерах нескольких мест использование обычных метеорологических наблюдений для получения данных, касающихся режимов диффузии.

Улучшены методы использования данных о флуктуации ветров для выведения коэффициентов диффузии и разработаны эмпирические методы, позволяющие применять обычные ветровые системы.

Описывается методика дымовой фотографии, дающая возможность осуществлять количественный анализ. Приводятся примеры применения в полевых условиях конкретных индикаторных методов для решения специальных проблем.

Одним из новейших метеорологических методов, применяемых при оценке площадок для размещения ядерных установок, является использование маяков-ответчиков на аэростатах для получения траектории движения воздуха и данных по диффузии на расстояниях до 160 *км*. Достаточно подробно описываются как сам метод, так и его возможности.

A/714 OMM

Recientes progresos en la aplicación de la meteorología a la seguridad de los reactores

por D. H. Pack *et al.*

Los recientes progresos de la tecnología nuclear han obligado a idear nuevos dispositivos y métodos meteorológicos que permitan determinar más exactamente en qué medida resulta satisfactorio el emplazamiento elegido para un reactor, a evaluar con mayor precisión la dispersión atmosférica y a medir las corrientes de aire; estas exigencias han sido plenamente satisfechas.

Cada vez es más neta la tendencia a elaborar programas meteorológicos con miras a responder a los requisitos de ciertos tipos de instalaciones nucleares y a resolver los problemas particulares que las mismas plantean.

Entre los nuevos métodos de elección del emplazamiento de los reactores que se describen en la memoria, se mencionan los análisis del gradiente vertical, efectuados sobre una base nacional, y los métodos que permiten determinar la persistencia de los regímenes del viento. Se explica, asimismo, con ejemplos, la manera de utilizar las observaciones meteorológicas clásicas para determinar los regímenes de dispersión en diversos emplazamientos.

Se han perfeccionado los métodos de aplicación de los datos relativos a las variaciones del viento en el cálculo de los coeficientes de dispersión y se han elaborado métodos empíricos que permiten aprovechar los regímenes clásicos de vientos.

Se describen en la memoria las técnicas basadas en las fotografías de humos, que facilitan la realización de análisis cuantitativos, y se dan ejemplos prácticos de estudios mediante indicadores que se prestan para resolver problemas especiales.

Uno de los recientes métodos meteorológicos aplicables a la elección de emplazamientos para reactores, consiste en utilizar instalaciones de radar transportadas en globo para determinar las trayectorias de las partículas de aire y conseguir datos sobre la dispersión en un radio de hasta 160 km. Los autores describen detalladamente este método y enumeran las posibilidades que el mismo ofrece.

Sûreté des piles de la filière uranium naturel-graphite-gaz

par R. Faÿs*, L. Laurent*, F. de Vathaire**, J. Plisson ** et J. Bussi ***

Le développement de la filière uranium naturel-graphite-gaz conduit à la réalisation d'ouvrages de tailles croissantes, depuis, pour la France, les réacteurs plutonigènes G2 et G3 de Marcoule, qui délivrent chacun 32 MW(e) sur le réseau national, jusqu'aux centrales EDF3 de Chinon et EDF4 de Saint-Laurent-des-Eaux, dont la puissance électrique atteindra 480 MW et qui seront mises en service respectivement au cours des années 1966 et 1968. Un palier ultérieur de 1 000 MW(e) est également possible.

Cette progression nécessite et justifie l'adoption de caractéristiques de plus en plus poussées, tant dans le domaine des performances imposées au cœur, que dans celui des dispositions .technologiques utilisées pour la réalisation du circuit primaire de refroidissement.

Les considérations de sûreté sont intimement liées au développement de la filière. Comme conséquence d'une étude théorique et expérimentale approfondie des divers accidents envisageables, les progrès dans la conception générale s'accompagnent également d'un renforcement de la sécurité.

Rappelons tout d'abord quelques facteurs contribuant à la sûreté intrinsèque de ce type d'installation:

a) Réactivité disponible à chaud limitée à quelques centaines de pcm (1 pcm = 10^{-5}), grâce au gavage de la zone centrale du cœur en combustible et éventuellement au moyen d'absorbants fixes. La vitesse et l'amplitude de possibles excursions de puissance sont ainsi réduites.

b) Coefficient de température de l'uranium négatif, d'où un effet stabilisateur agissant rapidement lors d'une libération accidentelle de réactivité. Le coefficient de température du modérateur devient positif en cours d'irradiation, mais agit avec une constante de temps de plusieurs minutes, suffisante pour laisser place aux corrections automatiques ou manuelles.

c) Division des absorbants neutroniques en de nombreuses barres de contrôle commandées par petits groupes (138 barres dans EDF4), ce qui limite les répercussions d'une anomalie affectant leur fonctionnement.

d) Faible section de capture neutronique du gaz carbonique; une baisse accidentelle de pression n'accroît pas la réactivité du cœur.

Le principal souci, lors d'un accident grave, est d'assurer le confinement des produits de fission radioactifs normalement contenus dans les éléments combustibles. Deux barrières principales s'opposent à leur migration vers l'extérieur.

La première est formée par les gaines étanches entourant les barreaux ou tubes d'uranium. Le circuit primaire, qui enferme le gaz carbonique sous pression, constitue la seconde. Il ne communique normalement avec l'atmosphère que par le système de réglage de la pression CO_2, dont les rejets sont contrôlés et filtrés.

Nous examinerons d'abord les problèmes liés au maintien de l'intégrité de ces deux barrières. Ensuite seront passées en revue les principales dispositions assurant la sécurité de fonctionnement en cours d'exploitation. L'analyse des accidents éventuels, de leurs moyens d'étude et des précautions prises pour en minimiser les conséquences, formera la dernière partie de l'exposé.

LES ÉLÉMENTS COMBUSTIBLES

Chaque degré gagné sur les températures de gaine et d'uranium permet des économies importantes, par augmentation du rendement thermique de l'usine et de la puissance spécifique dégagée dans le combustible. Toutefois, des considérations de sécurité s'opposant à une trop grande élévation de ces températures, un compromis doit être accepté, fonction des connaissances technologiques acquises précédemment dans les piles en exploitation et dans les installations d'essais, tel le réacteur PÉGASE du Centre d'études nucléaires de Cadarache.

Les éléments combustibles français actuels possèdent une gaine en magnésium, allié à 0,6% de zirconium. L'uranium contient 1,1% de molybdène (EDF3-EDF4). Nous nous limiterons à ce type de combustible.

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*** Groupement atomique Alsacienne Atlantique.

Comportement des gaines [1]

Du point de vue métalllurgique, le grossissement du grain limite présentement la température maximale de fonctionnement sur la surface externe à 470 °C environ. Aucune valeur inférieure n'est imposée pour des raisons de perte de ductilité ou de corrosion sous CO_2.

A partir de 450 °C se manifeste un noircissement superficiel par formation des composés MgO et CO_3Mg, accompagnés éventuellement d'un dépôt de carbone. Cette circonstance apparaît bénéfique lors d'un accident, car le refroidissement par rayonnement des éléments combustibles se trouve amélioré, même en l'absence d'échanges thermiques par convection.

La température maximale de fonctionnement à fixer dans le cas d'un réacteur déterminé peut être rendue inférieure à la limitation métallurgique ci-dessus, non seulement pour des raisons technologiques propres à l'installation (fluage des ailettes, comportement des bouchons d'extrémité, etc.), mais aussi, afin de ne pas dépasser, en cas d'accident grave, une température de sécurité.

Le point de fusion du magnésium et de son alliage au zirconium est de 650 °C. Par ailleurs, le risque d'inflammation dans le CO_2 dépend de la vitesse de chauffage et de la proportion de vapeur d'eau.

En atmosphère de CO_2 sec (teneur en eau inférieure à 160 parties en poids par million) l'inflammation se produit pour 940 °C (chauffage rapide) et pour 850 °C (échauffement avec stabilisation). Sous CO_2 humide (800 parties par million de vapeur d'eau), ces deux températures descendent respectivement à 800 °C et 650 °C. Ces résultats, obtenus avec une pression de 15 bars, illustrent l'influence très néfaste de la vapeur d'eau.

En atmosphère de CO_2, on peut conclure que la température de sécurité, à ne jamais dépasser, même localement lors d'un accident, est 650 °C, température du point de fusion, qui coïncide approximativement avec le point d'inflammation retardée en présence d'humidité. Toutefois, certaines hypothèses d'accidents peuvent conduire à une entrée d'air humide dans le circuit primaire. On adoptera donc finalement, comme température de sécurité, une valeur inférieure à la précédente, par exemple 620 °C.

Comportement de l'uranium

Le chauffage d'un alliage d'uranium à 1 % de molybdène jusqu'à 625 °C ne produit ni grossissement du grain pour les recuits de longue durée, ni changement dimensionnel pour un cyclage thermique rapide entre la température ambiante et 625 °C. Un passage rapide jusqu'à 645 °C ne détériore pas la structure. De 645 °C à 730 °C, un séjour de courte durée accroît la taille du grain, mais n'affecte pas les propriétés mécaniques (un cyclage thermique ultérieur crée une « peau d'orange » de 0,5 mm environ). Par contre, un long séjour doit être proscrit.

Par ailleurs, les effets de l'irradiation (croissance du grain, fluage Cottrel, fissuration), apparaissent peu dangereux dans le domaine des fortes températures, mis à part, éventuellement, un gonflement causé par les produits de fission.

Finalement, pour un alliage U-Mo 1 %, la température maximale métallurgique peut être fixée vers 600–625 °C (sous réserve d'une confirmation de la bonne résistance au gonflement, pour les forts taux d'irradiation). De courtes incursions au-dessus de 660 °C sont probablement tolérables.

Avec un combustible du type faiblement allié traité en phase β, le point chaud ne devrait en aucun cas dépasser 660 °C, température de transformation allotrophique $\alpha \rightarrow \beta$ pour l'uranium, sous peine de détruire l'effet de la trempe β et de restaurer un grain α grossier.

Des facteurs liés à la géométrie ou aux conditions d'emploi des éléments combustibles peuvent également limiter dans un réacteur donné, la température maximale de l'uranium à une valeur inférieure à la limite métallurgique ci-dessus définie.

L'ignition de l'uranium en présence de CO_2 n'advient qu'à une température élevée (860 °C pour un alliage U-Mo 1 %). Elle cesse lorsque le chauffage n'est plus assuré. Compte tenu, en outre, du caractère relativement paisible de la combustion, il ne paraît pas utile, lors des études d'accident, de considérer une température de sécurité pour l'uranium, contrairement à la procédure adoptée pour la gaine.

LE CIRCUIT PRIMAIRE

Une évolution sensible s'est manifestée, quant à la manière de réaliser l'enveloppe du cœur et de disposer les circuits d'échange thermique. Considérons successivement le cas des caissons en acier et celui des caissons en béton précontraint, conçus pour les centrales françaises.

Caissons métalliques [2]

Ce type d'enceinte fut adopté pour les tranches EDF1 et EDF2 de la Centrale de Chinon. Prenant comme exemple le caisson du réacteur EDF2, nous examinons brièvement le rôle des considérations de sûreté, aux différents stades historiques de l'ouvrage.

Le caisson d'EDF2 est une enceinte sphérique en acier au manganèse-molybdène de diamètre intérieur 18,3 m, percée de 8 ouvertures principales ø 1,6 m pour le passage du gaz carbonique. Pression de service: 26 bars environ. La température en partie basse (arrivée du CO_2) est de 190 °C, celle en partie haute (sortie CO_2) de 365 °C.

Période de conception et de construction

La détermination des contraintes fit l'objet de calculs théoriques, qui demandèrent à être vérifiés ou complétés par des essais sur modèles métalliques et sur des maquettes en plâtre ou araldite (par exemple, pour rechercher la forme et les dimensions optimales des renforts aux points d'aboutissement des tuyauteries).

Pendant le montage, des précautions spéciales furent prises pour éviter les contraintes de soudage (préchauffage ininterrompu à 200 °C, détensionnement ultérieur immédiat à 625 °C). Toutes les soudures ont été soumises à des contrôles radiographiques ultrasonoscopiques et magnétoscopiques.

Après achèvement de l'ouvrage, l'épreuve hydraulique s'exécuta avec une pression égale à 1,5 fois la pression de service, la température étant légèrement inférieure à 100 °C. Elle fut suivie d'une épreuve pneumatique dans l'air, sous 7 bars, à la température de 40 °C (sécurité en prévision des essais neutroniques à froid). Environ 600 jauges de contrainte étaient installées pendant ces épreuves.

Période d'exploitation

Dimensionnement et qualité des tôles et assemblages mettent l'ouvrage à l'abri d'une rupture ductile. Les chaînes de régulation sur la pression et les soupapes de sûreté interdisent tout dépassement du timbre. Par ailleurs, le fluage sous température ne se manifeste, pour les courtes durées correspondant à un incident éventuel, qu'au-dessus de 500 °C, ce qui laisse une marge de 500 °–365 ° = 135 °C.

Un risque de *rupture par fatigue* pourrait résulter de contraintes additionnelles apparaissant lors des phases de démarrage et d'arrêt, soit environ 240 évolutions, en admettant le rythme d'un arrêt par mois pendant 20 ans. Ce nombre étant faible, il ne s'agirait que de fatigue dans le domaine plastique. La région la plus sollicitée de l'ouvrage est la zone située entre les tuyauteries d'entrée et de sortie du CO_2, où la température passe de 190 °C à 365 °C. Le coefficient de sécurité est largement supérieur à 10 pour un tel nombre de cycles.

La sécurité par rapport à la *rupture fragile* résulte de l'impossibilité pour une crique d'apparaître et de se propager brutalement. L'épreuve hydraulique ayant été effectuée à 95 °C, cette température est prise comme limite initiale de sécurité pour la rupture fragile. Elle se situe entre les températures de la FTE *(fracture transition for elastic loading)* et de la FTP *(fracture transition for Plastic loading)*, alors que la température minimale de transition de ductilité (NDT) est d'environ 45 °C.

Lors de la mise en route du réacteur, on prévoit un préchauffage de l'installation à faible pression CO_2, afin de n'appliquer la pression de service qu'après obtention d'une température supérieure à 95 °C sur la totalité de l'enveloppe. Procédure inverse pour l'arrêt.

L'irradiation par les neutrons rapides durcit et fragilise l'acier, ce qui entraîne une augmentation de la température de transition, nécessitant une surveillance spéciale de la part des exploitants (essais de résilience sur des éprouvettes en acier, soumises dans le réacteur aux mêmes conditions de flux et de température que le caisson). En cas de montée de la température de transition, il y aurait lieu de hausser la température de consigne au-delà de laquelle est autorisée la montée en pression du caisson.

L'ensemble des précautions prises au cours de la construction, les résultats des épreuves hydrauliques et pneumatiques, le respect des consignes de mise en pression, l'examen ultérieur d'échantillons d'acier irradié, ainsi que le dépouillement des mesures de surveillance (dilatomètres, thermocouples, etc.) permettent de conclure à une sécurité d'emploi quasi totale de cette partie du circuit de gaz.

Aussi, lors de l'analyse des accidents graves, considère-t-on seulement l'éventualité de ruptures dans les diverses autres parties du circuit primaire (tuyauteries, compensateurs de dilatation, etc.), aussi peu probables qu'elles puissent d'ailleurs également apparaître.

Caissons en béton précontraint [3]

Déjà retenue en 1955 par le Commissariat à l'énergie atomique pour les réacteurs G2 et G3, où la pression CO_2 est de 15 bars, cette technique fut adoptée par Electricité de France à partir d'EDF3, qui fonctionne sous 26 bars. L'emploi du béton précontraint permet une augmentation des dimensions et de la pression de service des ouvrages, s'harmonisant bien avec les développements envisagés pour la filière.

De plus, EDF4 comportera, pour la première fois en France, la disposition appelée « à circuit primaire intégré », où échangeurs de chaleur et soufflantes sont placés avec le cœur, dans le caisson. L'ensemble du circuit primaire bénéficie ainsi de la grande sécurité apportée par la structure précontrainte.

Principaux facteurs de sécurité du béton précontraint

La tension de mise en précontrainte des câbles est supérieure à leur tension de service, par suite de la relaxation de l'acier, ainsi que du retrait et du fluage affectant le béton. La pose, à elle seule, constitue donc une épreuve pour l'armature.

De même, la situation la plus défavorable pour le béton apparaît, sous l'action des efforts de précontrainte et des gradients thermiques, lorsque l'enceinte se trouve hors pression, c'est-à-dire dans des conditions peu dangereuses pour le refroidissement du cœur.

Les câbles de précontrainte sont très nombreux (5 000 environ pour EDF4) et indépendants. Une perte de tension ou même la rupture de quelques-uns d'entre eux serait sans danger. De plus, l'injection d'un coulis de ciment peut réaliser une protection efficace des armatures contre la corrosion et contribuer à maintenir, après rupture, une certaine tension, par adhérence entre câble et béton.

Si, au cours du temps, la résistance du caisson devenait inférieure à la valeur escomptée, le premier symptôme visible, serait une fissuration de la face externe, causée par des contraintes thermiques de traction. On pourrait ainsi surveiller l'évolution du phénomène et agir en temps opportun.

En résumé, les caissons en béton précontraint présentent une remarquable sécurité. De plus, aucune rupture brutale n'est possible, l'ouvrage « prévenant » avant que puisse naître un désordre grave. Ces avantages s'étendent à l'ensemble du circuit primaire, lorsqu'une réalisation avec échangeurs intégrés entraîne la disparition des circuits CO_2 extérieurs, volumineux et relativement vulnérables.

SURETÉ EN EXPLOITATION [4]

Sources d'énergie

Dans toute centrale nucléaire, le secours est considéré comme absolu, lorsqu'il repose sur trois sources indépendantes pour alimenter les auxiliaires vitaux (appareillage de contrôle, soufflantes permettant le refroidissement à l'arrêt du cœur, etc.). D'où, par exemple, l'existence de trois groupes électrogènes Diesel à démarrage automatique. L'un d'eux étant supposé indisponible pour raison de panne ou de révision, les deux autres constituent, avec le réseau électrique général, un système de trois sources autonomes.

Pour les réacteurs avec circulation du gaz par turbo-soufflantes, et surtout lorsque la disposition générale exclut les phénomènes de thermosiphon (cas d'EDF4), une source de vapeur absolument sûre s'avère nécessaire: l'autonomie du réacteur ne dépasse une demi-heure environ, alors que le refroidissement doit s'étaler sur des durées beaucoup plus longues. Une solution consiste à prévoir une centrale à vapeur auxiliaire, comportant quatre chaudières indépendantes. L'adjonction d'une seule chaudière supplémentaire rend la centrale auxiliaire suffisante pour un site comportant deux piles. En outre, le traitement éventuel des fumées peut constituer un moyen de produire le CO_2 nécessaire au fonctionnement des réacteurs.

Dispositifs de sécurité

Enumérons les principaux:

1) La chaîne de sécurité entraînant une chute générale des barres de contrôle fonctionne à manque de tension électrique. Les principaux détecteurs qui la constituent sont triplés et raccordés selon le montage « au moins deux parmi trois ».

2) Des automatismes nombreux interdisent les fausses manœuvres, assurent la reprise en secours des organes vitaux et imposent une réduction de puissance en accord avec les possibilités du refroidissement.

3) L'étanchéité des éléments combustibles est vérifiée périodiquement, à intervalles de l'ordre d'une demi-heure, pour détecter les ruptures de gaine. Une surveillance permanente s'exerce ensuite dans le canal reconnu fuyard, jusqu'à son déchargement éventuel.

4) La régulation spatiale du flux neutronique, imposée, dans les réacteurs de grande dimension, par la nécessité de compenser les oscillations induites par l'effet combiné de l'absorption xénon et de la température, s'obtient par déplacements différentiels de certaines barres de contrôle, selon des critères de température. La comparaison aux seuils limites et l'élaboration des ordres correspondants est confiée, dans les centrales françaises, aux calculateurs numériques effectuant le traitement des informations.

Calculateurs numériques [5]

Les centrales nucléaires françaises utilisent largement les calculateurs numériques industriels, de type universel ou spécialisé, ce qui s'explique par la multitude des informations à traiter (détection de rupture de gaine, mesures analogiques, signalisations de type tout ou rien).

Les chaînes de sécurité et de régulation, sur lesquelles repose la sûreté proprement dite (puissance, températures de gaine, pression CO_2) sont indépendantes des calculateurs traitant l'information. Il n'est pas exclu que ces derniers élaborent des ordres intervenant dans le fonctionnement, mais il s'agit d'action ayant des conséquences limitées (modification des points de consigne de certaines régulations, par exemple).

De même, l'emploi d'un calculateur pour orchestrer le démarrage automatique de la centrale, solution envisagée pour les réacteurs EDF3 et EDF4, ne met pas en cause la sécurité, car les ordres ainsi envoyés ne concernent que de grandes séquences, possédant leurs propres circuits de verrouillage et de sécurité par relayages individuels.

Les calculateurs sont doublés. Ils effectuent périodiquement des tests internes de bon fonctionnement. En cas d'insuccès, la permutation normal-secours advient automatiquement, par exemple, après désamorçage d'un dispositif chronométrique extérieur, maintenu normalement excité au moyen d'une impulsion envoyée chaque minute par le calculateur en service.

Les équipements périphériques des calculateurs sont testés périodiquement par ces derniers, les plus importants d'entre eux étant également doublés.

ANALYSE DES ACCIDENTS

Les accidents graves examinés ci-après sont ceux capables d'entraîner la détérioration de l'une ou l'autre des deux barrières qui s'opposent à la diffusion atmosphérique des produits de fission. Il s'agit alors essentiellement de maintenir l'intégrité de la seconde barrière.

D'assez nombreux articles ont déjà été écrits sur le sujet, parmi lesquels [6] et [7] constituent de remarquables synthèses. Les accidents peuvent être répartis en diverses catégories, que nous examinerons ici successivement, dans le cadre des installations de Marcoule et Chinon [8].

Accidents de réactivité

Une libération incontrôlée de réactivité résulte généralement de la montée intempestive d'une partie des barres absorbantes. Compte tenu des différents facteurs de sûreté intrinsèque, l'évolution de la puissance thermique et des températures dans le cœur est toujours suffisamment lente pour que les capteurs correspondants de la chaîne de sécurité ordonnent l'arrêt de l'installation, avant tout danger. La figure 1 représente un tel accident pour EDF2.

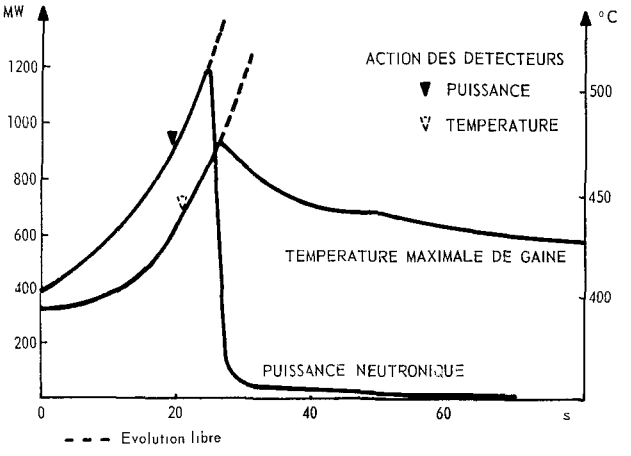


Figure 1. Accident de réactivité à demi-charge

Chute des barres par dépassement de température de gaine, après défaillance supposée de la sécurité sur la puissance neutronique

En période d'arrêt du réacteur, quelques barres de sécurité (4 dans EDF2-EDF3-EDF4) sont maintenues levées, prêtes à neutraliser une divergence intempestive, détectée par puissance neutronique exagérée ou temps de doublement trop court.

Accidents de soufflage

Les réacteurs de puissance comportent plusieurs circuits de refroidissement fonctionnant en parallèle, ce qui les protège contre un manque généralisé de soufflage. Des automatismes variés existent pour réaliser, lors d'un tel accident, les variations de puissance évitant toute répercussion dangereuse sur les températures des éléments combustibles. L'arrêt du réacteur peut également résulter des mesures de températures de gaines provenant des nombreux thermocouples judicieusement répartis dans le cœur.

Un manque de soufflage limité à un canal est nécessairement partiel, compte tenu de la grande surface de passage offerte au gaz et de l'impossibilité d'accéder normalement aux dispositifs de réglage du débit CO_2. Les canaux voisins évacuent une partie non négligeable de la chaleur produite. On bénéficie en outre de la forte capacité calorifique du graphite. De plus, toute atteinte à l'étanchéité du magnésium serait rapidement signalée aux exploitants par le système de détection des ruptures de gaines. Il n'existe donc pratiquement aucun risque de rejets radioactifs incontrôlés, ces derniers restant confinés dans le circuit primaire.

Accident de dégonflage

L'accident le plus grave pour ce type de réacteur consiste en une rupture brusque du circuit CO_2, susceptible a priori d'entraîner:

a) une désorganisation de l'empilement, par apparition de pressions différentielles importantes, empêchant le fonctionnement des dispositifs de sécurité.

b) un échauffement des éléments combustibles par manque de réfrigération, avec ruptures des gaines et émission de produits de fission en dehors du circuit primaire.

On élimine l'hypothèse d'une rupture de caisson, comme suite aux considérations développées précédemment. Pour un réacteur à circuit primaire non intégré, nous supposerons une rupture de tuyauterie principale CO_2 (diamètre intérieur 1,8 m dans EDF3), malgré sa faible probabilité d'occurrence. Examinons-en successivement les conséquences aérodynamiques et thermiques.

Effets aérodynamiques

Le cas le plus défavorable est celui d'une rupture de tuyauterie chaude, en partie supérieure des bâtiments. La décompression rapide du gaz contenu dans le volume surmontant le cœur risque d'entraîner le soulèvement des éléments combustibles, du graphite et des barres de contrôle. L'envol des éléments combustibles adviendrait d'abord, lors de l'augmentation de la pression différentielle entre les deux faces horizontales de l'empilement (cas des réacteurs à canaux verticaux).

L'étude théorique de ces écoulements gazeux repose sur une décomposition du circuit primaire en volumes

élémentaires, interconnectés par des résistances. Les pressions dans les volumes et les débits dans les résistances, sont déterminés à chaque instant par un système d'équations différentielles, résolues par calculateur numérique. Des perfectionnements à la schématisation peuvent être apportés pour simuler l'inertie du gaz dans les tuyauteries, ou même l'existence d'une capacité répartie le long des tuyauteries. Des essais sur maquettes, actuellement en préparation au Centre d'études nucléaires de Saclay, permettront d'apprécier la validité de ces représentations plus élaborées [9].

L'apparition de désordres dans le cœur, en cas d'accident de dégonflage, peut être supprimée au moyen des dispositions suivantes, toutes adoptées dans les centrales de Chinon:

1) Mise en place de tuyères ou de tubes Venturi dans les tuyauteries principales, à proximité du caisson, n'occasionnant qu'une faible perte de charge en service normal, mais réalisant par leur col conique une réduction de débit, lors d'une fuite importante (EDF1-EDF2).

2) Interposition de diaphragmes, dans les tuyauteries d'équilibrage des échangeurs qui communiquent avec le réacteur par plusieurs collecteurs en parallèle (cas d'EDF1).

3) Limitation du débattement possible des tuyauteries, ce qui permet une diminution du débit de fuite, par interaction des jets entre les deux lèvres d'une coupure affectant transversalement la section totale.

Effets thermiques

Un programme de calcul a été élaboré au CEN de Saclay, pour déterminer l'évolution des températures des matériaux du cœur après un accident [10]. Ce programme est utilisable également pour l'étude des accidents de réactivité et de soufflage. Le cœur s'y représente par un canal unique, comportant des couches annulaires successives (uranium, magnésium, gaz, graphite). Le bon accord entre la schématisation adoptée et la réalité des phénomènes fut vérifié lors d'expériences réalisées les 14 et 15 mai 1963 à Marcoule sur le réacteur G2.

Des expériences hors pile permettent l'étude de l'oxydation du graphite et de l'émissivité des gaines.

Pour un réacteur tel que ceux de Chinon, l'accident le plus défavorable au point de vue thermique, est la rupture d'une tuyauterie en partie basse, car le refroidissement du cœur diminue plus vite. Afin de limiter la montée en température des éléments combustibles, il importe d'introduire une forte antiréactivité dans les plus courts délais. A cet effet, certaines barres de contrôle sont munies de dispositifs d'accrochage sensibles au gradient de pression. Lors d'un dégonflage brutal, la barre se désolidarise de son treuil et chute rapidement.

A Marcoule et Chinon existent également sur le circuit gaz, des mano-contacteurs différentiels réalisant la chute générale des barres pour une variation brutale de pression. Le temps de réponse total de la chaîne est légèrement supérieur à celui des dispositifs de sécurité directe agissant sur l'accrochage des barres.

La figure 2 représente, à titre d'exemple, l'évolution des températures maximales de gaine et d'empilement dans le réacteur EDF2, pour un arrêt du refroidissement obtenu en 10 s. Au bout de 30 min., l'une des quatre soufflantes a été supposée remise en service, brassant du CO_2 à la pression atmosphérique. On constate que les plus fortes températures de gaines atteintes restent inférieures aux critères de sécurité définis plus haut et que le niveau général des températures dans le cœur baisse rapidement après mise en route du soufflage de secours.

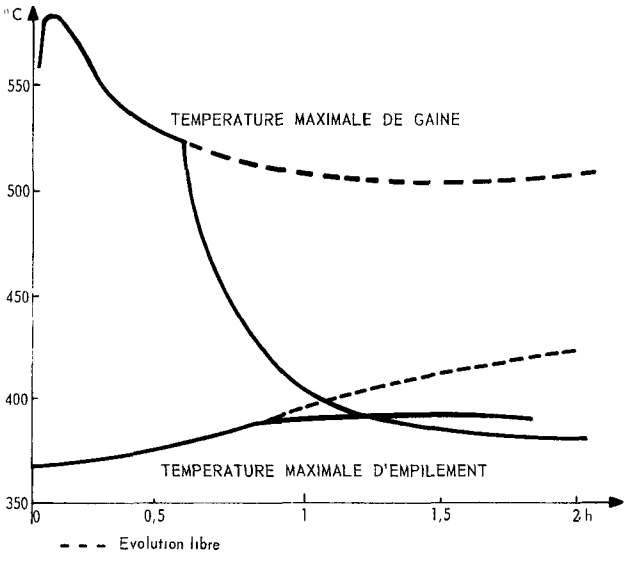


Figure 2. Accident de dégonflage

Une soufflante brassant du CO_2 à la pression atmosphérique est mise en service 2 000 s après la rupture

La présence d'air dans le CO_2 ferait apparaître le risque d'une oxydation exothermique du modérateur, favorisée par son élévation de température. Il importe donc de prévoir sur le site une réserve de gaz carbonique suffisante pour un balayage des circuits, capable d'éliminer l'air éventuellement introduit en début d'accident.

CONCLUSIONS

Deux barrières étanches, les gaines entourant l'uranium et l'enveloppe du circuit primaire, s'opposent normalement à la diffusion ambiante des produits de fission. La rétention dans le combustible constitue en outre, un obstacle supplémentaire.

Lors de l'accident le plus grave susceptible d'advenir (dégonflage occasionné par une rupture du circuit primaire) les températures maximales atteintes sur

gaines n'atteindraient pas les seuils correspondant à l'inflammation ou la fusion du magnésium, compte tenu des puissances spécifiques relativement basses dans ce type de combustible, de la grande capacité calorifique du graphite et de l'existence de moyens efficaces et sûrs pour effectuer le refroidissement en secours du cœur après la rupture.

Ces considérations justifient l'absence d'enceinte de sécurité étanche autour des réacteurs de la filière.

Pendant le fonctionnement, les impératifs de sûreté imposent des limitations aux températures des éléments combustibles. Les valeurs maximales sont déterminées à partir de thermocouples répartis dans l'empilement (sur les gaines et à la sortie CO_2 des canaux), compte tenu des singularités et des points chauds d'origines diverses, ainsi que de la déformation du flux neutronique, créée à faible puissance par l'enfoncement partiel des barres de contrôle.

Le maintien des températures est normalement assuré par les dispositifs de régulation ou par l'action des opérateurs. La chaîne de sécurité, qui provoque l'arrêt du réacteur constitue une troisième ligne de défense. Elle est réalisable de manière simple, essentiellement au moyen de détecteurs sensibles à la puissance, aux températures de gaine et à la pression CO_2. Elle ne doit pas présenter de partie commune avec les circuits d'automatisme et de régulation.

Les réacteurs de cette filière présentent donc une excellente sécurité, sans d'ailleurs qu'il en résulte une influence sensible sur le prix de revient des installations. Cette sécurité ne peut que s'accroître avec l'adoption de la disposition « intégrée » du circuit primaire, à l'intérieur d'un caisson en béton précontraint.

Toutefois, la sûreté d'un tel ensemble ne peut être considérée comme intrinsèque. Elle résulte en effet non seulement d'une judicieuse conception du projet et d'une parfaite exécution des montages et des essais de mise en service, mais aussi d'un strict respect des consignes d'exploitation et d'une surveillance attentive de l'évolution des propriétés des matériaux et du comportement des ouvrages, pendant toute la durée de vie de la centrale.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/81 France

Safety of natural-uranium-graphite-gas type reactors

By R. Fays *et al.*

Considerations of safety have played an important part in the development of French graphite-gas reactors, at present represented by the Marcoule and Chinon reactors, of which the EDF4 and EDF5 reactors are the latest stages.

Safety comes into play at both the design and the construction stage, particularly in studying the problems encountered in building the containment vessel enclosing the core and the primary cooling circuit (steel or prestressed concrete, external or built-in heat-exchangers).

During operation, safety mainly depends on judicious arrangement of the instrumentation in the core, on the satisfactory functioning of automatic devices and safety sequences, and on the control of any wastes that may arise. These various points are examined in turn. Safety requirements also call for the imposition of power ceilings, both in steady-state and in transient conditions.

The principal accidents that are liable to occur are reviewed. For the most important, the theoretical and experimental means available for studying them, and the steps taken to minimise their consequences, are discussed.

A/81 Франция

Безопасность уран-графитовых реакторов с газовым охлаждением

Р. Фейс *et al.*

Соображения по безопасности играют важную роль при разработке французских графитовых реакторов с газовым охлаждением, примером которых могут служить реакторы в Маркуле и Шиноне; следующим этапом разработки этих реакторов являются реакторы EDF-4 и EDF-5.

Соображения безопасности выступают уже в стадии проектирования и строительства установок для исследования проблем, связанных с сооружением оболочки для активной зоны и первичного контура охлаждения (сталь или предварительно напряженный бетон, внешние теплообменники или установленные в оболочке).

Во время эксплуатации безопасность основывается на разумном расположении приборов в активной зоне, хорошей работе автоматических механизмов и цепи аварийной защиты, а также на контроле возможных отклонений. Последовательно рассматриваются все эти различные проблемы. В целях безопасности необходимо также известное ограничение мощности в постоянных и переходных режимах работы.

Рассматриваются основные возможные аварии. Обсуждаются теоретические и экспериментальные методы, позволяющие изучить наиболее важные из мер, которые принимаются для уменьшения их последствий.

A/81 Francia

Seguridad de los reactores del tipo uranio natural-grafito-gas

por R. Fays *et al.*

Las consideraciones sobre la seguridad desempeñan un papel importante en el desarrollo del tipo de reactores de grafito-gas franceses, que está representado actualmente por los reactores de los emplazamientos de Marcoule y Chinon y cuyas etapas siguientes son los reactores EDF4 y EDF5.

La seguridad interviene en las fases de proyecto y de construcción, principalmente al estudiar los problemas relacionados con la construcción del recipiente que contiene el núcleo y del circuito primario de refrigeración (acero u hormigón pretensado, con los cambiadores de calor dentro o fuera del recipiente).

En la fase de explotación, la seguridad descansa concretamente en una disposición razonable de la instrumentación en el núcleo, en el buen funcionamiento de los automatismos y sistemas de seguridad, y en el control de los residuos. Se examinan sucesivamente estos diversos puntos. Los requisitos de la seguridad introducen igualmente limitaciones de la potencia en los regímenes estacionario y transitorio.

Se pasa revista a los principales accidentes que pueden producirse y se examinan los métodos, tanto teóricos como experimentales, que se utilizan para el estudio de los más importantes, así como las medidas tomadas para disminuir sus consecuencias.

La pile CABRI

par J. Ailloud et J. P. Millot *

CABRI est une pile expérimentale du type piscine qui est destinée essentiellement aux études de sûreté relatives aux piles à eau.

Au moment où les responsables des piles piscines et des réacteurs de recherche envisageaient l'augmentation de la puissance des réacteurs modérés à l'eau ordinaire, la Commission de sûreté des installations atomiques a reconnu la nécessité de la construction de CABRI pour permettre une évaluation plus réaliste des risques que comporte ce genre d'installations et une détermination plus exacte des marges de sécurité à respecter.

L'avant-projet de cette pile a été étudié par le Groupement industriel français atomique Alsacienne Atlantique (GAAA) sous le patronage de la Direction des piles atomiques du CEA.

L'organisation du programme expérimental et des études à effectuer avec la pile CABRI ainsi que leur réalisation ont été confiées à une équipe spécialisée du Département des études de piles appartenant au Service d'études de protection de piles.

Le Département de construction des piles fut chargé des compléments d'études nécessaires au projet et des travaux de construction proprement dits. Il a été assisté dans cette tâche par GAAA architecte industriel.

RAPPEL DES BUTS DE LA PILE CABRI

Le principal objectif de cette pile est de définir de façon aussi précise que possible les effets nucléaires, thermiques et mécaniques engendrés par des régimes transitoires, liés à des variations systématiques de conditions initiales déterminées, figurant des incidents ou des accidents sur les piles modérées à l'eau ordinaire.

A. – Il est prévu, essentiellement de réaliser des régimes transitoires consécutifs à des introductions de « réactivité initiale » soit en échelon, soit en rampe, et de les étudier expérimentalement en vue:

a) D'assurer les recalages avec les expériences similaires déjà effectuées sur SPERT.

b) D'examiner le comportement d'éléments combustibles. Cette étude est prévue par charge entière ou par élément. Dans le dernier cas, l'élément pourra être mis en capsule ou en boucle de façon à permettre une étude poussée jusqu'à sa fusion au moins partielle.

c) D'examiner le comportement des appareillages de contrôle et de commande de piles en général et l'incidence de ce comportement sur la sûreté de celles-ci. Pour cela, il est prévu d'éprouver principalement leurs qualités de stabilité de fonctionnement et leur temps de réponse pendant les montées rapides en puissance. Ces essais concernent les dispositifs tels que les chambres d'ionisation, les compteurs, les photomultiplicateurs associés ou non à leur scintillateur, les organes électroniques ou électromécaniques.

d) De définir les bases de recherches théoriques pour mieux comprendre les phénomènes d'excursion, en essayant de procéder par étapes successives: influence du coefficient de température, de la fraction de vide, des phénomènes d'ébullition, en tentant de préciser les limites au-delà desquelles les risques d'explosion deviennent non négligeables. Dans ces études, est examinée systématiquement l'influence des paramètres hydrauliques: hauteur d'eau au-dessus du cœur, température de celle-ci, sens de circulation (en particulier examen du passage de la convection forcée à la convection naturelle), débit.

Les phénomènes d'autolimitation de puissance mettent en jeu des facteurs physiques neutroniques et thermiques. Il est nécessaire d'étudier leurs évolutions conjointes dans le temps ainsi que leur mode d'interaction.

Pour les phénomènes neutroniques, dont les variations sont très importantes et très rapides, les méthodes de calcul et les théories habituellement utilisées doivent être revues et approfondies. Certaines études peuvent être qualifiées de fondamentales: ce sont celles qui ont trait aux équations elles-mêmes de la cinétique; il faut réétudier et certainement examiner de près ces équations pour déterminer la signification des paramètres utilisés: réactivité, temps de vie, fraction de neutrons retardés.

En effet, il est difficile de dire ce que signifient ces paramètres lors d'une variation importante et rapide de la réactivité, que cette variation soit due aux mouvements des barres de contrôle (ou de départ) ou à la contre-réaction liée aux phénomènes thermiques. Il pourra être nécessaire de reprendre la signification des paramètres à partir des données physiques fondamen-

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tales du cœur (équation de la diffusion); de même l'étude des problèmes posés par les cœurs comportant des éléments réflecteurs particuliers nécessitera de revoir et peut-être d'améliorer les formalismes déjà proposés.

D'autres études sont plus proches de l'expérimentation et visent à déterminer la valeur des paramètres qui interviennent dans les équations de la cinétique: réactivité, temps de vie, pourcentage de neutrons retardés. Celles-ci ont déjà été conduites suivant des méthodes qui sont connues mais qui devront éventuellement subir des modifications rendues nécessaires par les objectifs particuliers de CABRI. Demeurent également des problèmes tels que la détermination de la puissance en régime transitoire: évolution des pertes neutroniques en fonction de déformations transitoires éventuelles de la carte des flux, évolution des spectres de rayonnement gamma en fonction des neutrons retardés, variation de la sensibilité des appareils de mesure en fonction de l'énergie des rayonnements à détecter, temps de réponse des instruments de mesure eux-mêmes.

Pour les phénomènes thermiques, il est nécessaire d'entreprendre des études fondamentales sur:

a) La conduction calorifique transitoire dans l'épaisseur des éléments combustibles ainsi que dans le modérateur, ceci en vue de déterminer l'importance des dilatations dans le processus de contre-réaction et, en fonction de la période des montées en puissance, les instants de début d'ébullition ainsi que la fusion éventuelle de l'élément combustible.

b) L'ébullition en régime dynamique, c'est-à-dire, la vitesse d'apparition des bulles ainsi que leur évolution afin de voir clairement comment intervient la phase vapeur pendant des excursions de puissance et éventuellement son action dans les phénomènes de relaxation.

c) L'influence de la propagation calorifique par rayonnement.

d) L'évolution transitoire des pressions, des débits et plus généralement des phénomènes hydrodynamiques: relaxation de l'eau dans la cuve du cœur.

Les études appliquées dans le domaine de la mécanique et de la thermique sont essentiellement celles qui sont relatives à la signification des indications des différents instruments de mesures: pour la mesure des températures, recherche et évaluation de la perturbation locale apportée par l'instrument (thermocouple) et étude du temps de réponse et, pour les mesures de pression, étude des mêmes problèmes auxquels s'ajoutent les difficultés d'interprétation dues aux phénomènes d'oscillation propres aux capteurs.

B. – Il est prévu de réaliser des fonctionnements à puissance constante, pendant des intervalles de temps limités, en vue:

a) De préciser les marges de sécurité des piles piscines en régime d'exploitation en particulier celles qui ont trait aux éléments combustibles « burn-out »: la puissance maximale de fonctionnement liée à un accident de refroidissement, l'influence de l'ébullition locale sur le comportement nucléaire et sur les indications des instruments de contrôle, l'influence des phénomènes de cyclage sur la tenue dans le temps des éléments combustibles, etc.

b) De définir les bases de recherches théoriques sur les phénomènes d'instabilité à forte puissance, en régime de convection naturelle et en régime de convection forcée (par étude des fonctions de transfert).

c) D'examiner le comportement des appareillages de contrôle et de commande des piles en général, plus spécialement leurs performances à hauts niveaux (par exemple: raideur des dispositifs d'asservissement).

Ces études doivent être menées sur un certain nombre de configurations de cœur à réflecteurs variés et composées d'éléments combustibles de différents types. Là encore l'influence des paramètres hydrauliques, hauteur d'eau, température, sens de circulation, débit, sera examinée. Des expériences partiellement destructives sur un élément pourront être envisagées et différents essais devront permettre: d'une part de définir systématiquement les effets d'accidents causés sur une pile à eau, soit par erreur du personnel, soit par mauvais fonctionnement d'un dispositif de contrôle et de minimiser leurs effets et, d'autre part, d'examiner l'efficacité des dispositifs de sûreté et éventuellement d'en mettre au point de nouveaux.

Les travaux théoriques concernant ces études expérimentales se rapportent essentiellement ici à l'étude de la contre-réaction résultant de dilatations et de l'ébullition en fonction des conditions hydrodynamiques et du niveau de puissance (effet sur la réactivité, sur les temps de vie, sur les coefficients de fuite).

La principale étude appliquée consisterait à essayer de déterminer expérimentalement la contre-réaction en étudiant, par exemple, la fonction de transfert de la pile à différentes puissances en employant une méthode de modulation.

Chronologiquement, les études des régimes transitoires sont effectuées en premier lieu car leur réalisation n'exige pas les importants dégagements d'énergie que nécessitent, par contre, les études expérimentales des excursions de puissance qui impliquent des activations appréciables des installations proches du cœur de la pile.

INSTRUMENTATION UTILISÉE DANS CABRI

Nous n'envisageons ici que les mesures transitoires rapides. Les mesures effectuées en régimes permanents ou pendant des variations « lentes » de puissance sont effectuées par les appareils de mesures classiques des

autres piles. Nous diviserons l'instrumentation en deux domaines, nucléaire et non nucléaire.

Mesures nucléaires

Mesures globales de la puissance du cœur

Les appareils de mesures doivent:

a) Avoir une constante de temps aussi faible que possible (au plus quelques millisecondes);

b) Pouvoir couvrir une dynamique relativement grande (2 décades au moins dans le cas des chaînes linéaires);

c) Etre peu sensibles aux distorsions transitoires de flux dans le cœur (distorsions dues aux différents échauffements du combustible et du modérateur et éventuellement aux phénomènes d'ébullition). Ceci exige un appareil qui mesure un rayonnement peu atténué par l'aluminium et par l'eau; il s'agira donc d'un instrument placé à une certaine distance du cœur (de l'ordre de 1 m ou plus) mesurant la lumière de l'effet Cerenkov, le rayonnement gamma ou les neutrons émis par le cœur.

L'utilisation d'un détecteur à effet Cerrenkov nécessite des études et travaux préliminaires. Elle a été reportée à un stade ultérieur.

En ce qui concerne la mesure de la puissance à partir du rayonnement gamma, une étude expérimentale effectuée au Centre d'études nucléaires de Fontenay-aux-Roses [1] a montré que les perturbations dues aux gamma des produits de fission après le pic de puissance peuvent être très atténuées pourvu que la mesure soit effectuée à une distance du cœur supérieure à 2 m. En effet les gamma des produits de fission ayant une énergie inférieure aux gamma de fission et aux gamma de capture sont davantage atténués par l'eau. L'erreur provenant de la contribution des gamma des produits de fission reste inférieure à 1% dans ces conditions, la mesure étant effectuée avec une chambre d'ionisation. En utilisant un détecteur à impulsions (scintillateur associé avec un photomultiplicateur) il est possible de diminuer encore cette erreur systématique par une discrimination convenable.

La mesure de la puissance à partir des neutrons rapides est réalisée grâce aux neutrons thermiques qu'ils engendrent par leur ralentissement à proximité du détecteur. Ce dernier peut être soit une chambre d'ionisation à bore (mesure de courant), soit une chambre à fission ou un scintillateur gazeux associé à un photomultiplicateur.

Pour les mesures en courant, des essais effectués au Centre d'études nucléaires de Grenoble sur la pile Mélusine ont montré qu'une chambre à bore du type couramment utilisé dans le contrôle classique des piles (type CCC 2 B de la SACM par exemple) avait une constante de temps inférieure à la milliseconde pour un courant de l'ordre de 10^{-10} A au moins débité par la chambre.

En ce qui concerne les mesures par impulsions, en s'imposant une durée maximale de comptage de l'ordre de la milliseconde et en supposant une dynamique de mesure de 2 décades avec un nombre minimal d'impulsions pendant la durée de comptage de l'ordre de 100 (condition de précision des mesures), le pouvoir de résolution du détecteur et de son électronique associée doit être au moins de l'ordre de 10 nanosecondes.

Les détecteurs à scintillateur gazeux font actuellement l'objet d'études au Département d'études de piles, Service des expériences critiques.

En attendant leur mise au point, on utilise des chambres à fission subminiatures dont le pouvoir de résolution est de l'ordre de quelques centaines de nanosecondes seulement.

Mesures instantanées de flux en un point du cœur

Dans des expériences telles que l'étude de l'ébullition dans une boucle ou dans une capsule, l'étude de la fusion locale d'un élément combustible, etc., il est nécessaire de connaître aussi exactement que possible l'évolution du flux thermique en fonction du temps en un point particulier du cœur. Les qualités demandées au détecteur sont alors:

a) Etre aussi ponctuel que possible;

b) Pouvoir supporter une élévation de température atteignant 130 °C;

c) Avoir un temps de réponse de l'ordre de la milliseconde;

d) Fournir des résultats aussi précis que possible (quelque %) sur une échelle de puissance importante (2 décades au minimum);

e) Pouvoir fonctionner dans un flux de neutrons thermiques de 10^{16} neutrons/cm^2 s au minimum;

f) Pouvoir supporter des flux intégrés de l'ordre de 10^{15}.

Actuellement les seuls détecteurs qui puissent remplir ces conditions sont à notre connaissance les chambres à fission subminiatures (General Electric).

Mesures non nucléaires

Les mesures non nucléaires envisagées sont essentiellement des mesures de températures, de pression, de contrainte et de débit, effectuées en des points déterminés du cœur de la pile.

Mesure de températures

Nous avons voulu bénéficier des expériences déjà effectuées dans ce domaine sur les réacteurs SPERT plus particulièrement sur SPERT I et avons décidé d'utiliser des thermocouples similaires [2]. Les mesures sont

effectuées, en un premier temps, sur les faces extérieures des gaines des éléments combustibles.

La technique de fixation des fils de thermocouples utilisée est l'incrustation au moyen d'une soudeuse électrique par points. Il est prévu également d'utiliser une véritable soudure ponctuelle des fils conducteurs sur la gaine par fusion locale au moyen d'un éclair Laser.

Ultérieurement, des mesures seront effectuées à l'intérieur des éléments combustibles en constituant un thermocouple aluminium-nickel en profondeur dans l'élément. L'étude et la mise au point de cette technique sont en cours, sous contrat avec l'industrie privée. Parallèlement des études expérimentales sont menées hors pile pour mesurer les constantes propres aux thermocouples utilisés (temps de réponse et perturbation locale). Pour cela leurs indications transitoires sont comparées avec celles que donne une méthode de mesure de température complètement différente (phénomènes de photoluminescence) [3].

Mesures de pression

Compte tenu des informations dont nous avons bénéficié à partir des rapports publiés sur les piles SPERT nous avons décidé d'utiliser des capteurs à membranes dont on mesure la déformation grâce à une jauge de contrainte associée (capteurs type 4 313 et 4 316 de CEC). Compte tenu des phénomènes de résonance propre à ce type de capteurs, il nous a paru indispensable de faire réaliser des études systématiques dans ce domaine et un contrat a été passé dans ce but avec la Marine nationale à la Pyrotechnie de Toulon [4].

Mesures de contraintes

Les jauges utilisées sont du type étanche (Microdot et Baldwin).

Des essais de soudure électrique de ces jauges sont actuellement tentés sur l'aluminium et ne donnent pas entière satisfaction. Aussi, des essais de collage sont également prévus. La fixation des jauges sur les pièces en acier inoxydable n'a pas posé de problèmes particuliers.

Mesures de débit

Il est prévu d'effectuer des mesures de débit dans le pied de certains éléments combustibles du cœur au moyen de dispositifs à moulinet associé à une transmission électromagnétique. La gamme des mesures va de 0,5 à 25 l/s. Le temps de réponse est approximativement de 10 ms dans toute la gamme des mesures.

DESCRIPTION SOMMAIRE DE LA PILE ET DE SES INSTALLATIONS

Compte tenu du programme de fonctionnement [5] qui a été rappelé ci-dessus, la pile CABRI a été aménagée selon les dispositions suivantes:

1. L'implantation est réalisée dans une zone isolée du Centre d'études nucléaires de Cadarache;

2. La cuve du cœur est disposée en sous-sol dans une fouille creusée dans le calcaire et profonde de 10 m environ. Il a également été tenu compte au maximum des possibilités offertes par le relief du terrain pour obtenir des protections naturelles;

3. La pile proprement dite est contenue dans un bâtiment en forme de hall, du type renforcé (pression d'épreuve 40 cm d'eau) dont l'étanchéité est du type « amélioré » (passages étanches pour les câbles et les canalisations, porte d'accès à joint gonflable, etc.);

4. L'agencement des circuits de ventilation fait du bâtiment pile un ensemble à fuites éventuelles contrôlées par rejet après filtration dans une cheminée de 30 m de hauteur;

5. Autour du hall-pile, une zone d'exlusion d'un rayon de l'ordre de 300 m a été aménagée. Elle est matérialisée sur son pourtour par une barrière, l'entrée dans cette zone ne peut s'effectuer qu'en un seul point et elle est strictement réglementée;

6. Le contrôle de la pile — enregistrement des différents signaux, ordres de déclenchement, signalisations — se fait à distance depuis la salle du tableau de commande et celle qui est affectée à l'ensemble de centralisation des mesures dont il est question plus loin. Ces deux salles sont situées dans le bâtiment de contrôle proprement dit qui est reporté au-delà de la limite de la zone d'exclusion.

7. Au centre de la zone d'exclusion, outre le hall-pile proprement dit, se trouve une série de locaux formant un bâtiment extérieur qui lui est accolé et qui abrite:

a) La zone de ventilation comprenant: une salle de décompression de la nef-pile, une salle des ventilateurs et des compresseurs, une salle des filtres;

b) La zone constituée d'une salle de contrôle annexe qui contient des organes de répartition, les amplificateurs nécessaires pour les liaisons avec le tableau de contrôle proprement dit qui se trouve à 300 m, les dispositifs relatifs aux mesures transitoires nucléaires et non nucléaires (constitués schématiquement d'étages amplificateurs précédant un dispositif de commutation rapide pour le raccordement au centralisateur des mesures transitoires que nous verrons plus loin);

c) Une zone réservée aux installations électriques « courants forts »: cellules d'arrivée haute tension, protections des transformateurs d'alimentation, cellules basse tension de distribution, armoires de relayage, synoptiques des circuits hydraulique et électrique, local des batteries de secours associées à leurs redresseurs et à un onduleur prévu uniquement pour les circuits de contrôle;

8. Au-delà de la zone d'exclusion à 300 m de la nef-pile se trouve le bâtiment de contrôle, comprenant:

a) Une salle pour le calculateur et le centralisateur des mesures transitoires;

b) Une salle de contrôle proprement dite;

c) Une salle réservée aux expérimentateurs dans laquelle s'effectuent les contrôles des manipulations particulières;

d) Un laboratoire d'électronique;

e) Un laboratoire d'instrumentation transitoire;

f) Un laboratoire affecté à la Section de protection contre les radiations;

g) Une série de bureaux;

h) La chaufferie et les sanitaires;

9. Une ferme provençale placée à proximité du bâtiment contrôle a été rénovée et abrite en particulier les bureaux des théoriciens, une salle de réunion-conférence-documentation, un laboratoire de mécanique de précision, la salle des installations de comptage de détecteurs et le magasin uranium.

Descriptions particulières

Les différentes parties des installations qui méritent des mentions particulières sont les suivantes:

Cuve et support du cœur

Le cœur de la pile est contenu dans une cuve cylindrique de 5 m de diamètre et de 10 m de hauteur environ. Elle est du type auto-portante en acier inoxydable et peut supporter des surpressions de l'ordre de 40 m d'eau. Deux séries de hublots permettent l'observation des phénomènes ayant lieu dans le cœur et leur enregistrement au moyen de caméras (télévision ou cinéma).

Le cœur est constitué avec des éléments du type MTR (18 plaques, uranium enrichi à 90%). Ces éléments sont mis en place sur une grille comportant 10×10 emplacements. Dans chaque emplacement un dispositif à membrane élastique fonctionnant à l'eau sous pression permet de serrer les pieds des éléments combustibles et de les y maintenir dans le cas où un refroidissement par convection forcée dans le sens ascendant est utilisé (effort à l'arrachement de l'ordre de 150 kg).

La grille du cœur est supportée par un tabouret de 2 m de hauteur dont les pieds sont fixés directement sur le fond de la cuve du cœur. Ce tabouret supporte également la tuyauterie qui permet de faire circuler l'eau de refroidissement dans les éléments dans le sens vertical ascendant ou descendant. Cette tuyauterie est munie d'un clapet à verrou pour permettre aussi l'étude du refroidissement par convection naturelle.

Dans le plan horizontal de la grille du cœur, se trouve disposé également un certain nombre de grilles annexes destinées à recevoir les perches de contrôle et à faciliter la manutention des éléments et des dispositifs expérimentaux.

Circuit d'eau

a) Le circuit principal permet de faire circuler un débit maximal de 1 500 m^3/h dans le cœur par l'intermédiaire de tuyauteries en acier inoxydable. La vitesse de circulation de l'eau dans un élément combustible peut ainsi atteindre 5 m/s environ. Ce circuit permet, comme nous venons de l'indiquer, l'inversion du sens de circulation de l'eau dans les éléments.

b) Le circuit d'épuration et de remplissage utilise le principe d'épuration par résines échangeuses d'ions à lits séparés. Il permet de traiter au remplissage la totalité du volume d'eau nécessaire à la pile et aux circuits sans régénération intermédiaire. Ce même circuit sert également à l'épuration en circuit fermé et la résistivité de l'eau peut être maintenue entre 0,5 et 2 $M\Omega\ cm^2/cm$ pour un pH compris entre 5 et 7.

Ventilation

a) Lorsque la pile est à l'arrêt, son enceinte se trouve à la pression ordinaire, la ventilation et le conditionnement sont assurés avec recyclage partiel de l'air (en moyenne 1/3 d'air frais aspiré à l'extérieur à travers un simple filtre à poussière);

b) Lors des expériences, le bâtiment pile et ses annexes étant évacués, les portes fermées, l'étanchéité réalisée, on crée à l'intérieur du hall-pile une dépression de l'ordre de 4 mm d'eau: une aspiration est réalisée par un ventilateur d'extraction asservi avec un organe déprimomètre et les gaz extraits sont dirigés sur des filtres « absolus » incombustibles et sur piège à iode avec rejet dans une cheminée de 30 m de hauteur.

Barres de contrôle-sécurité et barre de départ

Le cœur de la pile CABRI comporte cinq barres: quatre barres de contrôle-sécurité et une barre de départ.

Les barres de contrôle-sécurité sont des barres absorbantes d'un type similaire à celles qui équipent des piles telles que MÉLUSINE, TRITON et SILOÉ. Elles sont dotées chacune d'un mécanisme qui permet de les déplacer verticalement dans le cœur avec un repérage de leur position à $\pm$ 0,2 mm près et d'un dispositif de sécurité à propulsion pneumatique pour leur réintroduction brutale dans le cœur (en 0,1 s environ).

La barre de départ est constituée d'un élément absorbant semblable au précédent équipé d'un dispositif pneumatique permettant de l'éjecter rapidement vers le haut hors du cœur (en 0,1 s environ) à partir d'une cote réglable repérée avec la même précision que les barres de contrôle.

Contrôle transitoire

En plus des appareillages de mesures conventionnels qui constituent l'équipement de contrôle et de commande d'une pile classique, CABRI est équipée d'un ensemble d'appareils qui permet d'effectuer un certain nombre d'enregistrements pendant la durée d'une expérience: mesures nucléaires de flux, mesures de températures, mesures de débit, mesures extensométriques de contraintes et de pression, etc. à concurrence de 48 voies.

L'enregistrement des différentes mesures a été prévu, en première étape, sous forme arithmétique par la méthode de l'échantillonnage périodique sur chaque voie de mesure. Pour cela, on utilise l'organisation logique et les mémoires d'une machine électronique à calculer (RW330) qui sert ainsi de centralisateur des mesures. Les possibilités de cette machine sont utilisées pour réaliser également: des télécontrôles et des télécommandes avant et après expériences, des calculs de traitement d'information sur les résultats bruts des différents échantillonnages de façon à obtenir sous forme de tableaux ou de courbes sur une table traçante, des résultats exploitables plus commodément et directement par les expérimentateurs. En fonction des expériences à réaliser, il est possible de faire varier la cadence de scrutation sur les voies de mesures; la cadence de base est de 400 par voie et par seconde (durée 2,5 ms pour un cycle de scrutation complet des 48 entrées).

Il est possible d'obtenir une fréquence d'échantillonnage plus élevée sur des voies déterminées en raccordant celles-ci à plusieurs entrées régulièrement espacées sur le dispositif de scrutation. Il est possible au contraire de ralentir la cadence en bloquant périodiquement l'enregistrement des mesures sur chaque voie, celle-ci continuant à être explorée à la cadence de base. La capacité de mémoire du centralisateur de mesures permet d'effectuer des enregistrements, à la cadence de base, pendant 6 s au maximum.

En deuxième phase, il est prévu d'effectuer les enregistrements d'un certain nombre de voies de mesures sur bandes magnétiques en utilisant le procédé de modulation de fréquence. Le centralisateur de mesures sera alors équipé des organes nécessaires pour l'analyse des bandes magnétiques ainsi obtenues. La durée d'enregistrement des phénomènes sera alors portée à 15 min. environ.

Premiers résultats expérimentaux

La divergence « classique » de la pile CABRI a été effectuée le 21 décembre 1963 et la première excursion transitoire de puissance a été réalisée le 16 avril 1964.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/103 France

The CABRI reactor

By J. Ailloud and J. P. Millot

It became necessary to build in France a reactor which would make it possible to study the operating conditions of future atomic energy plants, and to select, test and perfect the safety devices to be used in them. A water-water reactor of the same general type as the latest laboratory or university reactors to be built under the auspices of the French Atomic Energy Commission was decided upon: it seemed desirable to be able to evaluate the risks involved and to study the possibilities of providing the increases in output constantly demanded by users; and clarification of the phenomena of power oscillations and the risk of burn out proved to be of special interest.

The work programme of CABRI will be harmonised with work carried out on the US SPERT reactors of the same type; while CABRI was being built, very useful contacts were established with the US specialists who designed these reactors.

CABRI is an open-core, swimming-pool reactor, without lateral shielding, housed in a reinforced building with controlled leakage at the Nuclear Research Centre at Cadarache. It stands alone in the middle of a prohibited area 600 m in diameter. The control and metering gear are located at the periphery of the zone. CABRI uses MTR fuel elements. The control and scram rods are operated by compressed air. The maximal flow in the coolant circuit is 1500 m^3/h. Transient measurements are recorded by a RW 330 unit.

CABRI is intended for:

(*a*) Studies on the safety of water-water reactors, comprising:

(i) the definition of safety margins under operating conditions: research into the maximum power at which a swimming-pool reactor can be allowed to operate in the light of the possibil-

ities of coolant accidents; effect of local boiling on the nuclear behaviour of the reactor; performance of the control and safety gear and instruments in limiting conditions, and so on;

(ii) experimental studies of power excursions linked with specific initial conditions. The aim of this work is to define the basis for theoretical research, and the limits beyond which the risk of explosion ceases to be negligible. It will be carried out in such a way as to enable the results to be checked against those obtained in reactor experiments elsewhere, and with the object of continuing the experiments into the explosive region; and

(iii) studies of the behaviour of reactor control and operating devices;

(*b*) Experimental investigations into transient conditions of short initial duration (study of boiling, and measurements of temperature, pressure and vacuum) with the aim of defining the bases for a theory of swimming-pool reactor kinetics related to heat-transfer problems; and

(*c*) Studies of the behaviour of fuels in reactors (it is planned to carry out theses experiments in loops).

CABRI went critical on 21 December 1963, and the first transient power excursion took place on 16 April 1964.

A/103 Франция

Реактор CABRI

Ж. Эллю, Ж. П. Милло

Во Франции возникла необходимость построить реактор, который позволил бы изучать условия работы будущих установок, выбирать, испытывать и уточнять системы аварийной защиты для этих установок. В связи с этим был выбран водяной реактор, относящийся к числу новых проектов Комиссариата по атомной энергии Франции в области реакторов для лабораторий и университетов. Действительно, необходимо оценить, какую опасность они представляют, и изучить возможности увеличения мощности, постоянно требуемого потребителями. Особый интерес представляет выяснение явлений колебания мощности и опасностей пережога.

Программа работ на реакторе CABRI будет сходна с работами, которые проводятся на американском реакторе SPERT такого же типа; во время строительства реактора были установлены плодотворные контакты с американскими специалистами, которые первыми разработали такие реакторы.

Краткое описание реактора

CABRI является погружным реактором с открытой активной зоной, без боковой защиты. Он установлен в Центре ядерных исследований в Кадараше, в здании с защитой и контролем утечек. Он находится в центре запретной зоны радиусом 300 *м*. На периферии этой зоны устанавливаются контрольные и измерительные приборы. Тепловыделяющие элементы реактора относятся к типу используемых в реакторе MTR. Регулирующие и аварийные стержни приводятся в действие сжатым воздухом. Максимальный расход контура охлаждения составляет 1500 $м^3$/ч. Измерения в переходных режимах мощности регистрируются на установке RW-330.

Назначение реактора и программа работ

Реактор CABRI предназначается:

а) для исследований по безопасности водяных реакторов: 1) уточнения пределов безопасности работы, т. е. изучение максимальной мощности, при которой может безопасно работать погружной реактор в зависимости от потери теплоносителя, влияние местного кипения на поведение реактора, изучение работы регулирующих и аварийных устройств в критических условиях; 2) для экспериментального исследования разгонов мощности, связанных с начальными условиями; их целью является определение основы для теоретических исследований а также определение пределов, выше которых нельзя пренебрегать опасностью взрыва; исследования будут проводиться таким образом, чтобы обеспечить возможность сопоставления с опытами вне реактора, которые должны их продолжить в направлении возможности взрыва; 3) для изучения поведения оборудования, предназначенного для управления и контроля;

b) для экспериментальных исследований, касающихся переходных режимов при коротких начальных периодах работы (изучение кипения, измерение температуры, давления и пустотного коэффициента), с целью определить основы теории кинетики водяных реакторов, связанные с явлениями теплопередачи;

c) для изучения поведения топлива в реакторе (эти эксперименты намечено провести в петлях).

Первые результаты экспериментов

Реактор CABRI введен в эксплуатацию 21 декабря 1963 года. Первые испытания в переходных режимах мощности проведены 16 апреля 1964 года.

A/103 Francia

El reactor CABRI

por J. Ailloud y J. P. Millot

Fue necesario construir en Francia un reactor que permitiera estudiar las condiciones de funcionamiento de instalaciones futuras con el fin de elegir, probar y poner a punto los dispositivos de seguridad que hayan de ser adoptados. Se ha elegido un reactor de agua, tipo que corresponde a las construcciones más recientes del CEA en lo tocante a reactores de laboratorio o de universidad; efectivamente, conviene poder evaluer los riesgos presentados y estudiar las posibilidades del aumento de potencia pedido constantemente por los usuarios; es particularmente interesante aclarar los fenómenos de oscilación de potencia y los riesgos de quemado *(burn-out)*.

Los programas de trabajo con el CABRI se armonizarán con los trabajos efectuados con los SPERT americanos del mismo tipo; en el momento de su construcción se establecieron contactos fructíferos con los especialistas americanos que primero dieron a conocer estos reactores.

El reactor CABRI es de tipo piscina con núcleo abierto, sin protección lateral, instalado en el Centro de Estudios Nucleares de Cadarache en una edificación reforzada y de fugas controladas. Está situado en el centro de una zona de exclusión de un radio de 300 m. Las instalaciones de control y de medidas están situadas en la periferia de esta zona. El reactor emplea elementos combustibles del tipo MTR. Las barras de control y seguridad están impulsadas por aire comprimido. El caudal máximo del circuito de refrigeración es de 1500 m^3/h. Las medidas transitorias se registran en un conjunto RW 330.

El reactor CABRI está destinado a:

a) Estudios de seguridad de reactores de agua:

i) Para precisar los márgenes de seguridad en funcionamiento: estudios de máximo de potencia al cual es prudente autorizar el funcionamiento de un reactor de piscina en función de un accidente de refrigaración, de la influencia de la ebullición local en el comportamiento de los instrumentos de control y de seguridad en condiciones límites.

ii) Estudios experimentales de excursiones de potencia ligadas a condiciones iniciales determinadas: su objeto es definir las bases de investigaciones teóricas y los límites fuera de los cuales los riesgos de explosión dejan de ser despreciables. Estos estudios serán dirigidos en forma a asegurar acoplamientos con experiencias exteriores al reactor, capaces de extenderlos hasta el dominio explosivo.

iii) Estudios del comportamiento de los aparatos de control y mando del reactor.

b) Estudios experimentales referentes a los regímenes transitorios de cortos períodos iniciales (estudio de la ebullición, medidas de temperatura, presión y fracción de vacío, etc.) con el fin de establecer las bases de una teoría sobre la cinética de reactores de agua ligada a los fenómenos de transferencias térmicas.

c) A estudios sobre el comportamiento de combustibles de reactores (estas experiencias están previstas en lazos cerrados).

El reactor CABRI se hizo supercrítico el 21 de diciembre de 1963 y la primera excursión transitoria de potencia se realizó el 16 de abril de 1964.

Reactor safety analysis as related to reactor siting

By F. R. Farmer *

THE DEVELOPMENT OF REACTOR SAFETY CRITERIA

The nuclear industry is the first industry in the world that has tried to assess its hazards *a priori* and control its risks in advance. This has had two results: first, that in the early stages it developed quite elaborate techniques for this purpose, and contributed a large staff effort to the specific problems of safety in design and in operation; secondly, that it has been a remarkably safe industry from the start. So far, over 300 reactors have been built and operated in various parts of the world; 40–50 of them are large power reactors with a total capacity of over 5 000 MW(e) (i.e., equal to the entire installed capacity of the UK in 1930). The safety record is excellent; certainly there have been a few minor mishaps and a small number of larger plant failures well-publicised, but only in exceptional cases have a few people closely associated with the plant received harmful doses of radiation, and the author is not aware of a single case in which members of the public have been exposed excessively to radiation.

This record has been achieved at considerable cost in money and skilled manpower and it must be maintained; but there is scope for constantly reviewing safety methods to see whether they can be simplified without any depreciation of standards—perhaps even with some improvement.

Safety criteria are beginning to emerge in the form of guides, recommendations and licence requirements. As these criteria can profoundly influence the development of reactor systems they are important not only to safety, but to reactor technology as a whole on account of their impact on the economics of nuclear power. It is imperative, therefore, to get safety criteria right. Well-judged safety requirements, besides promoting safety, will evoke an active response from designers and operators, and thus will help in the development of new, improved and more economic systems. Ill-chosen, over-complex or out-of-date safety requirements will do neither effectively: they will frustrate the development of a reactor system without really promoting safety; not carrying conviction, they will tend to be evaded and will create a false presumption of safety.

In the early stages of development, when few data are available, some safety criteria must be arbitrary and inevitably over-cautious. Only too easily, however, they can then become stereotyped if the essential principles and the rules originally determined by convenience are not clearly distinguished and if they are not critically and imaginatively re-examined quite frequently.

Many countries are interested in procedures which might assist the siting and safe operation of reactors. This fact, and the growing diversity of reactor types, make the present an opportune time to consider the merit of various practices which are now emerging. Ideally, this should be done impartially without regard to reactor type or national sponsorship, as the promotion of safety must override national or commercial interests. The following comments on some of the basic concepts and present practices are offered in this spirit and should not be read as showing preference for any particular type of reactor.

REACTOR SAFETY ASSESSMENTS

In order to operate a reactor safely, it is necessary to explore the effects of minor disturbances which are likely to occur during normal operation. These require extensive study, and confirmation of the assumed values of the important parameters during the commissioning and subsequent operation of each reactor. This work forms a major part of safety assessment, but it is often overshadowed by the spectre of major accidents of extremely remote probability.

The consequences of such major failures are explored theoretically in such safety assessments, to show either that fission products are not released in large quantities from the fuel or that such a release can be controlled by containment or other means. However, the variety of accident conditions and their various combinations present an insuperable problem in analysis. As any one problem is studied, some new refinement is suggested or some other coincident or consequent condition postulated. Accordingly, to avoid this endless step by step approach, a few minor and major faults are selected to span the range of

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possibilities and provide a background on which decisions may be made as to:

(*a*) Site and licence conditions;
(*b*) System or component improvements;
(*c*) Operating limitations;
(*d*) Emergency procedures.

THE "MAXIMUM CREDIBLE ACCIDENT" CONCEPT

Many in government and industry who are concerned with these decisions need to know the results of accident studies, but few have the time, and few the specialised knowledge and training, to read and critically appraise a full safety report. This has led to a search for some yardstick of reactor safety and to the adoption of the "maximum credible accident" as a concept which might be evaluated, expressed as a number and used in formal procedures. It is the author's view that whilst this approach may have some limited value in the comparison of the safety of reactors of the same type or family, its use to compare the safety of reactor systems of different types is meaningless.

A comparison by this method between a typical gas-cooled reactor (GCR) burning Magnox-canned uranium fuel, and a water reactor of either the boiling (BWR) or pressurised (PWR) type illustrates this contention. For consistency with common practice in the use of this concept, this study is conducted as follows:

(*a*) Failure of the primary pressure circuit is chosen as the basis of the "maximum credible accident". Further faults which increase the severity of the consequences are progressively added and explored. These are:

(i) Assume double-ended fracture of one pipe or duct in the primary circuit, but otherwise reasonably predictable behaviour of equipment and fission products;

(ii) Select the position and size of the fracture to give maximum inconvenience
or
apply pessimism to the behaviour of auxiliary equipment or fission products;

(iii) Apply a further *arbitrarily* imposed condition of failure to fuel or protective devices;

(iv) Add further pessimism.

(*b*) Fission product release is quoted in terms of the release of ^{131}I from a core inventory of 10^7 curies [equivalent to a reactor power of 500 MW(th)].

The results of this study are set out in Table 1.

The tabular presentation implies that the computed releases can be used for comparison of the safety of two reactors. But plainly this would only be valid if the fault conditions leading to those releases were equally probable in each system. This is hardly likely to be the case although the use of the common title of "maximum credible accident" to describe two different sets of failures imports such an implication. In fact, it is obvious that the fault conditions selected have little in common. However carefully these have been chosen, they remain largely arbitrary. The fact that an arbitrarily chosen fault in one system may lead to a release greater than that from a different arbitrarily chosen fault in another system cannot be translated

Table 1. A Selection of "maximum credible accidents"

GAS-COOLED REACTOR	WATER REACTOR
(1) *Double-ended pipe fracture* Maximum can temperature 20 °C below limit NO FISSION PRODUCT RELEASE	(1) *Double-ended pipe fracture* Partial melting of core releases 10^2 curies Deposition reduces to 10^1 c. Containment closures effective Leakage 1 : 1 000 per day RELEASE (1 DAY) 10^2 C
(2) *Fracture in selected position of area* [a] 0.7 *to* 0.8 M^2 Probability 1/100 of reaching temperature limit on hottest can NO FISSION PRODUCT RELEASE	(2) *Full core melting or no plating* Assumptions introduce one factor of 10 in pessimism RELEASE (1 DAY) 10^3 C
(3) *Assume one channel fails* Partial retention of iodine in reactor and filtration system FISSION PRODUCT RELEASE 10^2-10^3 C	(3) *Assume slight deterioration in containment; effective opening* 1.0 cm^2 Leakage rate high $\approx$ 10% per day RELEASE 10^4-10^5 C
(4) *Assume several channels on fire* RELEASE INDETERMINATE	(4) *Assume failure of valve closure or missile penetration of containment* RELEASE INDETERMINATE

[a] The implication of selecting a precise area and position of fracture is discussed elsewhere [1].

into a comparison of the safety of the two. More especially is this so if the probability of occurrence of those faults, or the speed or manner of their development, is very different in the two cases.

Additional major faults could be postulated to demonstrate the point still further. For example, a coincident failure to shut down the GCR will result in extensive fire, whereas the neutron power of the water reactor is automatically terminated by moderator boiling. Conversely, some reactivity faults in a water reactor are induced by a rapid change in density of the moderator—through cold water injection or collapse of voids. These may require an immediate insertion of control rods whereas a similar condition cannot arise on a GCR.

These are not the only reasons to doubt the intrinsic worth of the numerical answers obtained by these methods. The numerical results in Table 1 for conditions (1), (2) and (4) of the GCR case, of either zero or indeterminate value, are unusable as aids to site selection or emergency procedures. Even where positive numerical answers are obtained, these are far from precise and, at best, indicate no more than a probable order of magnitude for the release. Yet, siting criteria such as those relating to distances (as used in the US [2]), or those relating to population numbers (as used in the UK [3]), are sensitive to factors as small as 2.

It appears then that the "maximum credible accident" concept is unlikely to provide a meaningful approach to the comparison of over-all safety of reactors of different types. Not only is this so at the present time when comparisons are made on a wholly theoretical basis but, since major faults will rarely occur, experience will not accumulate sufficiently for the concept to be applied on a factual basis.

The several well-known reactor systems currently used for power production have all been carefully developed with great attention to their safety. Examples are the BWR, PWR, GCR, and heavy water tubular systems. Each of these systems has its own special safety features for which equal merit might be claimed, even when very high standards of prudence are applied. Thus the commercial success of those systems is unlikely to be influenced by these various claims. In the author's opinion, the dissimilarity of the systems defeats any attempt to place them in any real order of safety, and, subject to reservations arising from the particular design and execution of any specific reactor proposal, all the basic types, as used for power production, have justifiable claims to equality in siting.

THE VALUE OF MAJOR FAULT ANALYSES TO REACTOR SAFETY

Although the "maximum credible accident" approach to the safety comparison of reactors appears to have little merit, it is important to recognise the value to reactor safety of major accident studies, especially in the important matters of identifying key parameters and critical features, of inspiring appropriate safety research, and of promoting developments which improve intrinsic safety or improve defensive mechanisms. These consequential benefits are amplified and exemplified as follows.

A theoretical study of a major accident and its consequences will normally identify those parameters (such as control rod worth, drop times, relevant reactivity coefficients, instrument reliability, delay times, etc.) which are important to the safety of the reactor. Such identification then enables arrangements to be made for the values of those parameters to be verified factually both initially and as frequently as is necessary throughout the life of the reactor.

The theoretical analysis of major faults usually reveals a few particular features which would critically determine the course and consequences of an accident. For the GCR, two such features are the insertion-time for the control rods and the initial temperature distribution of fuel cans. A reduction of the delay time has been achieved on stations of the Central Electricity Generating Board by a fast acting release fitted to a selected number of control rods [4, 5]. An intensive effort in all UK GCRs. has been applied to a conservatively based statistical analysis of fuel can temperature [4-6]. This work has improved the operator's knowledge of his reactor and reduced some areas of uncertainty in the accident analysis. This, together with improved equipment, has enabled him to operate the reactor more efficiently and at higher power.

The early studies of hypothetical accidents revealed uncertainties in the response of the reactor to fast transients; in the mechanism of the release of fission products from fuel and their subsequent behaviour; and in the modes of failure of pressure systems. The identification of these and other problems gave impetus to safety research programmes in many countries, directed to two major objectives, first of reducing the probability of a major fault, and second of improving defensive measures to reduce the consequences of an accident. Much progress has already been made; notable examples in the first category are investigations of the effect of irradiation on steel [7], of the fracture mechanics of steel pressure vessels [8], and by means of model tests of transient pressure distribution [9]. Fundamental and applied research in fission product behaviour [10], on the other hand, and investigations of the effectiveness of containment, have been directed to the second objective of improving defensive measures.

Well-judged safety requirements—derived from fault studies—evoke an active response from designers and promote the development of new and improved systems. These incorporate the results of research and embody features which tend to reduce liability to fault, or effect improvements in defensive systems.

The use in the advanced gas-cooled reactor of relatively inert oxide fuel clad in stainless steel offers advantages in terms of safety as also does the recent introduction of concrete pressure vessels in France and the UK. It is generally held that these structures cannot fail catastrophically and consequently that the reactor cannot undergo a rapid depressurisation. Whilst this conclusion must be the subject of detailed evaluation for each design, the potential gain in safety has been accepted in the UK and reactors so designed benefit from a relaxation in site rating [11].

The trend in some recent designs of gas-cooled and water reactors is to an integral system in which substantially the whole of the primary circuit lies within one pressure vessel. This significantly reduces the probability of fast failure and, with further improvements in design and progress in research, it may be shown that such failures are virtually impossible.

Major developments in the defensive protection offered by containment have been cited in recent proposals to locate water reactors near population centres. These rely on improved containment by using a double-walled "zero" leakage shell, or by reducing pressure by vapour suppression techniques. Other developments have improved the reliability and performance of filters and absorbers used in contained or vented reactors to retain a high proportion of those fission products of biological significance.

These then are some of the values of major fault analyses. Since reports dealing with such analyses are widely discussed, and since selected major faults are used in regulatory procedures, the sponsor may feel pressure to achieve greater precision in the analysis of consequences and to undertake extensive supporting research programmes. It should be recognised that, for the purposes of safety comparison, a search for a precise statement of the consequences of a few arbitrarily selected faults is unjustified if it obscures the greater uncertainty about the initial liability to fault and the pattern of its subsequent development. Nevertheless, the exercise in major fault analysis as a whole can promote better understanding of the reactor and as already indicated, can lead to the identification of key parameters and critical features and to the promotion of research. It is with these objectives that relevant parts of the study will require great precision and careful selection.

THE ESTABLISHMENT OF SITING PRINCIPLES

The problem of assessing the relative safety of a reactor is complex and difficult. But some method has to be devised so that siting and operating requirements may be tackled logically. In such a situation there is need for vigilance to avoid being driven by pressures of administrative and legislative convenience to produce readily manipulated formulae or rules against which reactor safety is to be tested. Once this happens the efforts of designers and operators are directed to compliance with just those requirements, and further thought on the effect of their efforts on real and fundamental safety may well be neglected. Indeed, such formulae or rules, framed as much by the dictates of convenience as anything else, may themselves quickly become mistaken for essential safety criteria.

For instance, reactor types could be placed in an order of preference by some judgment which, however honest and well-intended, would remain largely arbitrary. If this order were then adopted in the framing of requirements governing reactor siting and safety, then by a circular process subsequent comparisons by tests against those requirements would only sanctify the original judgment.

When any country chooses to establish a nuclear power programme and selects a reactor system, it thereby commits itself as an "act of faith" to the future safety of that system and its development. It is unlikely that experience will confirm or confound the wisdom of this choice before siting decisions have to be made. If the programme calls for a number of sites, these will have to be appropriately distributed in relation to the power distribution networks of the country so that the costs of transmission will at least be tolerable. In densely populated countries such as the UK and many other European countries and Japan, these requirements alone limit the degree of acceptable isolation and, therefore, narrow the choice of the areas available for reactor siting. The sites chosen initially for the UK programme represent a balance between isolation and the available areas as decided by those requirements.

In this way, initial siting principles are established which are appropriate to the specific features and problems of a particular country, and which provide a base line for the whole power reactor programme. In the formulation and application of these principles, numerical methods may be used to classify environmental populations, as, for example, those used in the USA [2] and the UK [3] and others of a similar nature.

THE GROWING IMPORTANCE OF COMPARATIVE ANALYSIS IN REACTOR SAFETY AND SITING

As the programme advances, it will be necessary to re-appraise the bases of reactor safety and siting to take account of new information from research or experience (though, because of their very infrequency, it is unlikely that major faults will contribute to this experience). There will also be a need to judge the relative safety of subsequent and more highly developed versions of the same reactor type. This can best be

done on the basis of a comprehensive safety assessment and not merely a study of the consequences of a few selected major faults; although such a study would be an integral and important part of the whole.

Reactor safety and siting is, of course, only one of the problems of a developing nuclear power programme. The choice of a reactor system in one of its forms of development, and the direction of reactor operation both involve technical considerations on which the government of a country will seek guidance from its own experts, possibly with assistance from elsewhere. A full safety assessment requires the skills and experience of a group of engineers and scientists who should severally be familiar with the analytical techniques employed, with recent and relevant developments in physics, mathematics, engineering, metallurgy and instrumentation, and who should be abreast of the techniques, practices and experiences of nuclear plant operation thoughout the world.

Although judgment based on such assessments must reflect to some extent the personal preferences of the individuals concerned, there will in fact be a large measure of agreement between experts in any country (and indeed between experts of all nationalities) on those features of any particular reactor which can significantly affect safety. These assessments cannot be expected to give answers in precise numerical terms which would permit easy and direct comparison of reactors, but they do enable credit to be given for advances in safety, either by reducing still further the slight degree of risk originally accepted or by relaxing siting and operational requirements. In this way there is a continuing incentive for research into safety aspects and improvement in design.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/182 Royaume-Uni

Analyse de la sûreté des réacteurs en fonction de leur emplacement

par F. R. Farmer

L'une des nombreuses techniques employées pour estimer la sûreté d'un réacteur consiste à considérer une certaine série de défaillances hypothétiques, pour en étudier les conséquences et en particulier pour déterminer s'il se produit un échappement de produits de fission et, s'il en est ainsi, de quelle amplitude.

Un de ces accidents hypothétiques est souvent appelé accident « maximal croyable » et l'échappement de produits de fission en résultant, exprimé sous forme d'un seul nombre, acquiert alors une importance exagérée et tend à cacher les impondérables à la fois dans la définition et dans l'analyse.

L'utilisation des analyses de défaillances de réacteur d'une manière routinière établit une familiarité excessive avec des concepts tels que « accident de rupture de conduite » et « rétention », qui prennent alors une signification dont la précision n'est pas justifiée.

Le mémoire illustre l'imperfection de ce genre d'analyse, explore les difficultés et suggère la nécessité de maintenir une attitude critique à l'égard de l'analyse des défaillances et une certaine flexibilité dans les règles utilisées concernant l'implantation et l'autorisation d'utilisation des réacteurs.

A/182 Соединенное Королевство

Анализ проблемы безопасности реактора, связанной с выбором строительной площадки

Ф. Р. Фармер

Одна из многих методик, используемых при оценке безопасности реактора, заключается в постулировании определенного ряда гипотетических аварий, в изучении их последствий и особенно в определении возможности высвобождения продуктов деления и в случае наличия такой возможности — в определении их активности.

Одна из таких гипотетических аварий рассматривается иногда как «максимально возможная», и последующее высвобождение продуктов деления, выражаемое определенным числом, затем приобретает преувеличенное значение и скрывает неясности при определении и анализе.

Использование какого-то установленного метода анализа аварии реактора ведет к чрезмерно вольному обращению с такими понятиями, как разрыв трубопровода и удержание выброса внутри защитной оболочки, что требует тогда неоправданно точного определения значений.

В данном докладе иллюстрируется недостаточность такого подхода, исследуются различные трудности и предполагается необходимость придерживаться критического подхода к анализу аварий и добиваться гибкости руководств по выбору площадки для реактора и при выдаче лицензий.

A/182 Reino Unido

El análisis de la seguridad de dos reactores en relación con su emplazamiento

por F. R. Farmer

Una de las muchas técnicas utilizadas al estimar la seguridad de los reactores consiste en postular una serie determinada de averías hipotéticas, estudiar sus consecuencias y, sobre todo, determinar si tiene lugar o no el escape de los productos de fisión y, en caso afirmativo, la magnitud de aquél.

Tal accidente hipotético se identifica a veces como el « máximo probable » y el escape correspondiente de productos de fisión, expresado con un simple número, adquiere entonces una importancia exagerada y encubre las incertidumbres de la definición y del análisis.

La utilización de un esquema rutinario en el análisis de los fallos del reactor presupone el conocer a fondo conceptos tales como el « reventón de un tubo » y la contención, que adquieren entonces, sin justificación, un significado preciso.

La presente memoria demuestra lo inadecuado de este método, explora las dificultades y sugiere la necesidad de analizar las averías con espíritu crítico y de ser flexible en la interpretación de las guías utilizadas en la autorización y en el emplazamiento de los reactores.

The underground containment of the Lucens experimental nuclear power plant

By E. Binggeli, P. Verstraete * and A. Sutter **

The Swiss experimental nuclear power plant at Lucens, some 30 km north of Lausanne, is at present under construction and located underground. It will be equipped with a 30 MW(th) pressure-tube reactor with slightly enriched uranium, moderated by heavy water and cooled by carbon dioxyde. The nuclear part of the plant, the turbo-alternator and the installations for maintenance as well as the fuel storage pond will be located in three distinct caverns while the control room, laboratories, Diesels, workshop and offices will be housed in the service building (Fig. 1). Underground construction is particularly suited to Swiss conditions and it has often been adopted for the construction of hydro-electric plants. The experience thus gained and the inherent advantages from the point of view of safety have played a decisive role in accepting this construction technique for the first nuclear power plant in Switzerland.

The rock strata covering the reactor cavern constitutes a natural barrier of protection against the effects of a possible nuclear accident. Containment analysis will, therefore, be worked out in such a manner as to profit most from such conditions. Tests on the site and in laboratories have shown that the porous rock at Lucens could be used to retain and store contaminated gases that would escape after a severe accident. Rock seepage water which could become contaminated during such operation would be collected in a drainage system and channelled to the station for treatment of liquid radioactive waste.

REACTOR ACCIDENTS

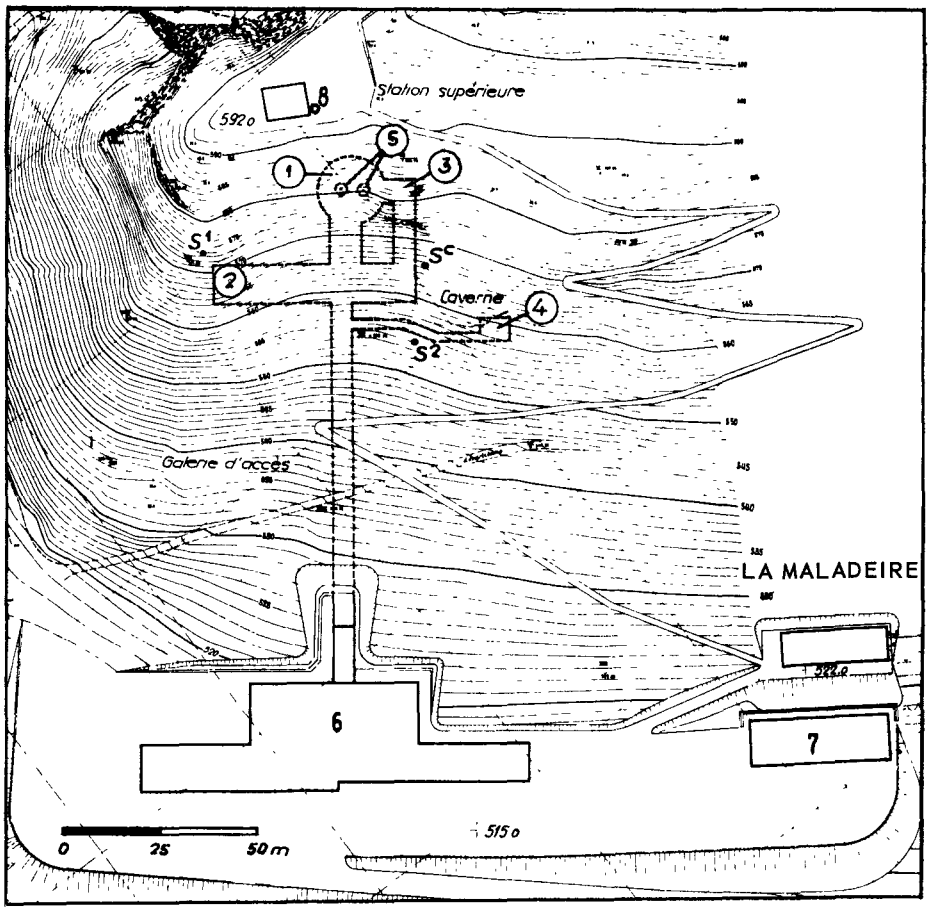
Plant design provides adequate protection against most equipment faults and it is only in the case of a major failure in the safety system or following an accident of extreme severity that, in order to protect the surrounding population, it would become necessary to utilize the actual containment. The walls of the reactor cavern and the surrounding rock mass constitute the containment which is designed to withstand the load of the maximum hypothetical accident, defined by the complete rupture of the thermal circuits, melt-down of all fuel elements including energy release from exothermal chemical reactions [1]. Although an accident of such severity is not deemed possible, it is, however, retained for containment analysis. Following such a hypothetical accident the overpressure in the reactor cavern would rise to a maximum of 2.3 atmospheres while the temperature would reach 110 to 525 °C according to the extent of damage to the secondary steam circuit and the release of chemical energy.

DESCRIPTION OF THE CONTAINMENT

The rock at Lucens is a molasse of sedimentary geologic configuration, with horizontal stratification and very few fissures or faults. The rock shows a very high porosity and a low permeability. In the rock mass, thick layers showing a course texture are alternated by generally thin layers of very fine rock texture. The reactor cavern is a cylinder some 30 m high and 18 m in diameter covered by a rock mass of more than 40 m. A layer of very porous concrete is cast adjacent to the rock and will constitute a drainage system intended to collect seepage water. The inner wall of reinforced concrete will support the intermediate floors and the structures in the reactor cavern. A bituminous coating consisting of 2–3 layers of 0.1 mm thick aluminium foils imbedded in bitumen sandwiched between these two concrete walls give the cavern the required leak-tightness. It is important to note that this lining envelopes the entire reactor cavern independently of the inner structure. The access to the cavern is sealed by a large steel wall comprising the two air locks and the material hatch as well as penetrations for all pipings and cables. A large steel frame anchored to the concrete will hold the steel wall. At the interface between concrete and steel, leak-tightness is achieved by allowing the bitumen coating to overlap the steel frame where it will be clamped by a flange.

* Bonnard et Gardel, Lausanne.

** Atomelectra Ltd., Zurich.



1. Reactor cavern
2. Turbine hall
3. Fuel pond cavern
4. Experimental cavern
5. Vertical test shafts
6. Service building
7. Cooling towers
8. Ventilation building and stack

S^1, S^2, S^c Test borings

Figure 1. General site layout at Lucens

CONTAINMENT PRINCIPLES USING THE PROPERTIES OF POROUS ROCK

Following a severe accident heat transfer to the walls, inner structures and equipment will cause a rapid fall in temperature and pressure in the reactor cavern. After approximately half an hour, the temperature would be below 100 °C and the overpressure will not exceed 0.3 atmospheres, this residual pressure being sustained by the mass of non condensible carbon dioxyde released during the accident [1]. Furthermore, this slight overpressure would assist heat transfer in the core thus limiting the possible extent of melt-down of fuel.

An important contamination will derive from short-lived gaseous fission products which decaying would yield long-lived radioactive solides retained by filters. Considering these various phenomena, it becomes apparent that emergency action should be initiated only approximately half an hour after the accident. After this time, owing to the slight overpressure in the cavern and the inevitable leakage, pressure will continue to decrease very slowly. Leakage, from the containment constitutes an inconvenience only in as much as it is channelled towards the other adjacent caverns where airborne activity could accumulate to concentrations not acceptable for direct release to the atmosphere through the 50 m ventilation stack at the top of the hill. Such considerations and the urgency to suppress the potential danger from a cavern of more than 4 000 m^3 filled with highly contaminated gases under pressure

lead to the necessity of relieving this overpressure quickly after the accident. In the case of Lucens this can be achieved by expanding the contained gases into the surrounding porous rock. Accordingly, two distinct phases can be considered after the accident. During the first phase, temperature and pressure will decrease through cooling effects. Most of the short-lived radioactive gases will have decayed to solids while condensing vapour will contribute to the wash-out of radioactive contaminants. The second phase will start only about half an hour after the accident and will call for the controlled decompression of the cavern. To this effect, pressure relief valves penetrating in the containment walls will be remotely opened releasing radioactive gases to the surrounding porous rock. Due to the drainage concrete the gases will be distributed immediately and uniformly around the cavern and under the driving force of the overpressure, will gradually flow into the rock. The filters installed before the relief valves will provide filtration of particulate contaminants.

CONTAINMENT CRITERIA

In order to provide adequate protection of the surroundings, the following conditions pertinent to the type of containment described must be satisfied:

(*a*) Permeability of the rock must be sufficient to permit the flow of gases whilst the porosity of the rock must be sufficiently high to allow storage and retention of these gases in the vicinity of the cavern. Furthermore, the rock mass must be very homogeneous and exempt of fissures.

(*b*) Seepage of rock water must affect the entire rock mass and occur sufficiently slowly to allow continuous and adequate decontamination in the waste treatment plant following an accident. Furthermore, rock water which will not be collected in the drainage system of the various caverns must flow sufficiently slowly favouring partial decay of contaminants before reaching the surface.

(*c*) Particulate contamination resulting from the decay of rare gases must be filtered and retained by the rock.

(*d*) The leak-tightness of the containment must be sufficiently high so as to retain the major portion of radioactive rare gases and halogen vapours during the first phase after the accident. Leakage to the neighbouring caverns must be restricted to such an extent as to permit normal ventilation in these caverns and exhaust to the atmosphere within the limits of permissible radioactive burden.

(*e*) The reactor cavern must be constructed to withstand the effects of temperature and pressure that may result from any severe accident.

PRINCIPAL RESULTS OF TESTS CONDUCTED FOR THE CONTAINMENT ANALYSIS

Air permeability of the rock at Lucens has been measured on the site, particularly in the shaft (Fig. 1) excavated along the vertical axis of the reactor cavern and in a specially equipped experimental cavern. At various points, probes were introduced into the rock where the local permeability could be measured by recording the rate of pressure decay in a standard reservoir connected to the probes. Air flow into the rock causes an exponential decay of the initial pressure in the reservoir so that a relaxation time τ (reduction by factor e) can be assessed depending on the local nature of the rock and varying proportionally to the insertion depth d measured between the point of the probe and the free surface of the rock. Thus d/τ is a relative measure of local gas permeability in the rock. All measured values of d/τ are within the limits of 4.08 to 12.9 cm/min for rock with coarse texture and between 0.09 and 0.15 cm/min for rock with very fine texture. Absolute values for rock permeability were evaluated using a similar technique with water and taking into account the difference in viscosity of air and water. The corresponding coefficients of permeability according to Darcy are of the order of 7.3×10^{-7} m/s for coarse rock and of 5×10^{-9} to 8×10^{-9} m/s for fine rock.

The flow velocity of water in the rock was determined first by recording the effects of forced circulation over a short distance and then by measuring the effects of natural circulation over a large distance.

(*a*) In the first test a water head was established between a test boring of some 70 m depth and the experimental cavern 3 m away, causing forced circulation. By pumping water into the boring throughout the experiment the water level could be maintained at 35 m above the natural level of rock water in this boring. The water was initially labelled with sodium chloride with a concentration of 10 g/l. The free rock surface (coarse rock) of the adjacent cavern which was initially dry became soaked some five minutes after the beginning of the test and a constant flow-rate was observed and recorded thereafter. Precipitating Cl-ions with Ag NO_3, first labelled water could only be detected 29 hours after the beginning of the test. This implies that under pressure resident rock water is quickly displaced but contamination is very slow. Considering the porosity of the rock and the pressure gradient established between the source and the measuring point yielding a flow velocity of 0.1 m/h, a value of the coefficient of permeability of $6.6 (\pm 1.4) \times 10^{-7}$ m/s could be established. This value was used as a basis to calibrate the values of relative permeability measured at various points in the rock using the pressure decay method described previously.

(*b*) The second test consisted in observing the natural flow of rock water labelled with fluorescein and iodine. This water was introduced in the test boring. Sc and was tested at various points in the existing excavations as well as at various source points at the surface of the hill. At an interval of one month, the two tracers were added to the water, i.e., 5 kg of fluorescein giving an initial concentration of 10 g/l, followed by 13 kg of iodine corresponding to an initial concentration of molecular iodine of 100 g/l (I_2 dissolved by adding 20 kg of KI). With the help of fluoroscope, fluorescein labelled water could be observed for the first time 241 days ($\pm$ 17) after the beginning of the test and the presence of iodine was detected by starch test some 311 ($\pm$ 12) days both at the same point 20 m away from the source. From these tests one could conclude that the maximum flow velocity of rock water is of the order of 30 m/year. Furthermore, the direction of the observed natural flow indicates that after a severe accident most of the contaminated water would in fact be collected in the drainage system around the reactor cavern. Moreover, water escaping collection in the drainage system around the cavern would flow for some three years in the rock mass before reaching the surface of the hill, leaving a sufficient time for radio-iodine to decay completely (^{131}I, $t = 8.05$ days). These tests, especially the second one, pertaining to the whole of the excavations, confirm the homogeneity of the rock and the absence of large and extended fissures which were anticipated, based on previous geologic investigations.

(*c*) The flow rate of rock water seeping into the excavated reactor cavern was measured to be 5 m^3/day independent of rainfall.

(*d*) An order of magnitude of the time of residence of water in the rock during the time between rainfall and detection was estimated by determining the tritium content of the rock water. Tritium content from water probes collected in summer 1962 at a point corresponding to the bottom of the reactor cavern is equivalent to 10 TU, which shows that this water entered the soil in the form of rain before the explosions of the 1954 H-bombs after which tritium content of rainwater now varies between 50 and 200 TU (1 TU, tritium unit = 10^{-18} atoms T/atoms H). These results also confirm the very slow flowing velocity of rock water.

Laboratory tests show that the rock porosity defined as total void to total volume varies between 12% for fine grain rock and 27% for very coarse grain rock. In order to determine the fraction of volume available for gas retention, the natural water content in the rock has been measured, showing that all lower layers around the reactor cavern and the upper layers with fine rock texture are saturated with water, while rock showing a course texture is only partly wetted. This pattern has been confirmed by results obtained in the experimental cavern which show that the effective porosity available for gas retention varies between 6 and 12%.

The effect of the rock on radioactive contaminants

Laboratory tests show that molasse acts as a reducing agent on molecular iodine in water solution. Due to the relatively high water content in the rock, radioactive iodine which would be introduced into the rock after gas expansion would first dissolve and then be reduced to ionic state. Iodine content in the rock would be too minute to condense. However, in contact with the molasse, it would be reduced to very soluble iodine. This mechanism will essentially be responsible for potential contamination of underground water following a severe accident.

Absorption and wash-out of strontium ions in the molasse were investigated. Absorption was simulated by introducing finely ground rock into a solution of Sr^{++} after which the absorbed quantity of strontium was determined indirectly by titration with a flame photometer. For initial concentrations of 500, 1 000 and 1 500 mg of strontium per litre, the absorbed quantities of strontium were 1.5, 3 and 5 mg per gram of rock. Wash-out effects on absorbed strontium were simulated by passing a solution of strontium nitrate through a column of finely ground molasse followed by water wash-out. Strontium content in this water is shown in Fig. 2 for three typical rock samples. These

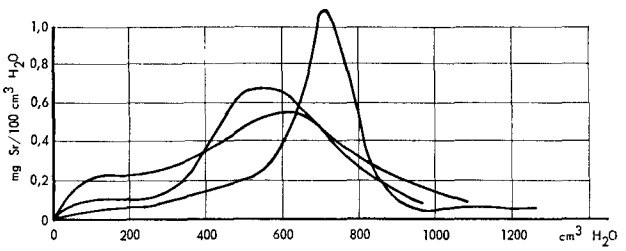
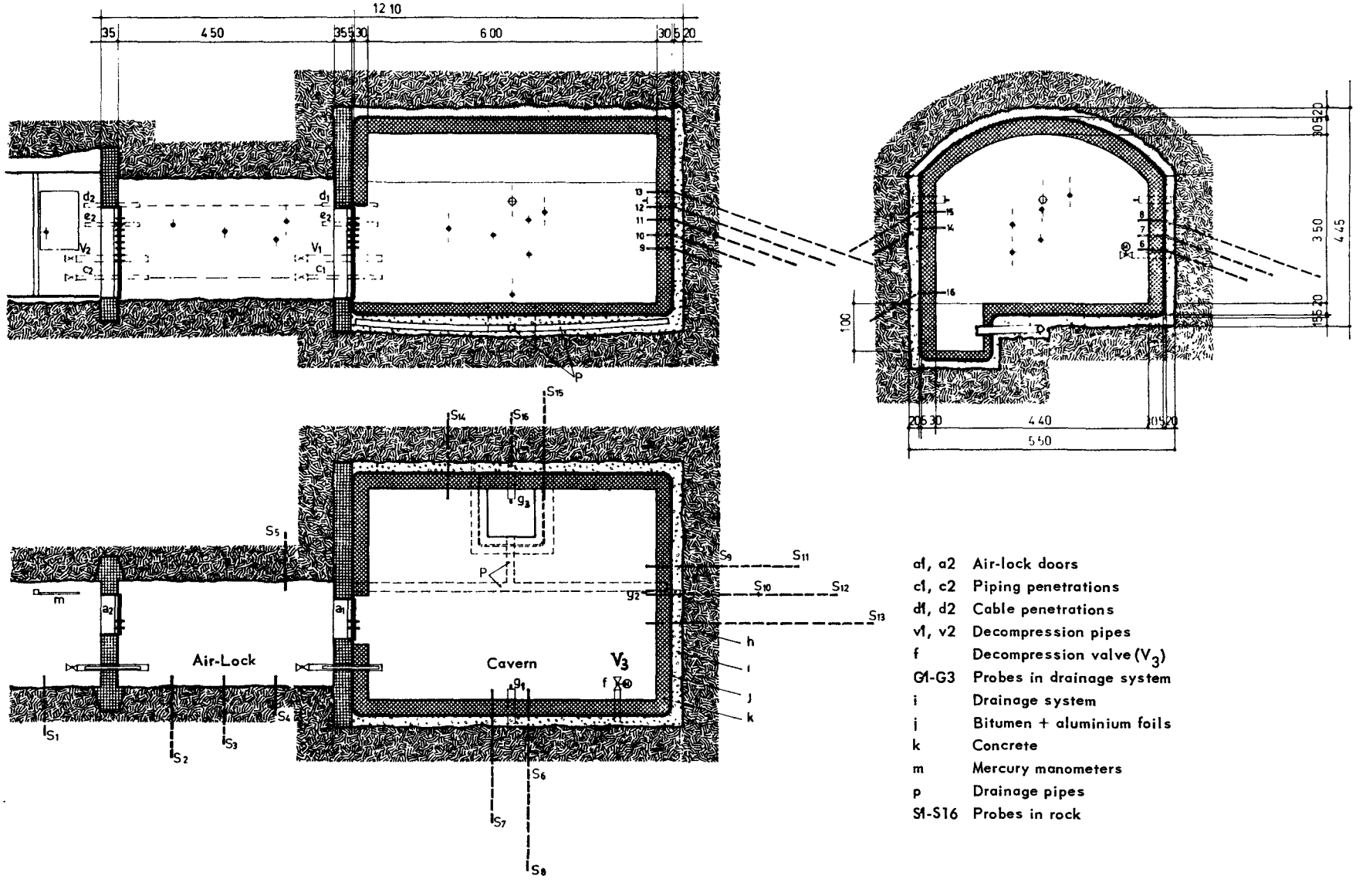


Figure 2. Absorption and wash-out of strontium in typical molasse

results show that no durable strontium contamination can subsist in the rock, since all the initial strontium content passed to the wash-out water. Due to the alternating affect of absorption and wash-out in the rock the progression of strontium would in fact be at least ten times slower than that of water.

Experimental cavern excavated on the site

The main task of this cavern was a small-scale simulation of construction methods envisaged for the large containment in order to verify the possibilities for expansion and retention of gases in the rock following an accident.



a1, a2	Air-lock doors
c1, c2	Piping penetrations
d1, d2	Cable penetrations
v1, v2	Decompression pipes
f	Decompression valve (V_3)
G1-G3	Probes in drainage system
i	Drainage system
j	Bitumen + aluminium foils
k	Concrete
m	Mercury manometers
p	Drainage pipes
S1-S16	Probes in rock

Figure 3. General layout of the experimental cavern

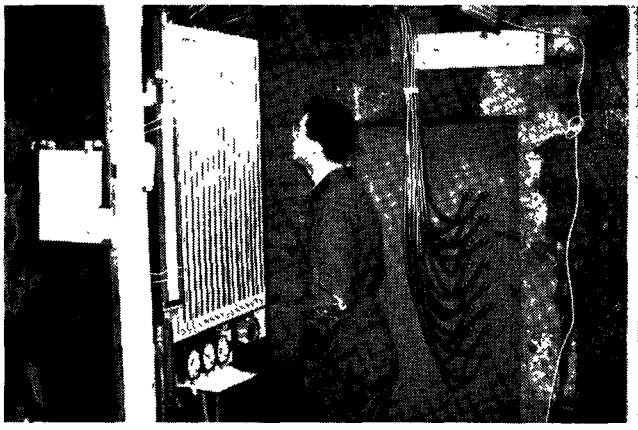


Figure 4. Manometer panel with probe connections through air-lock door

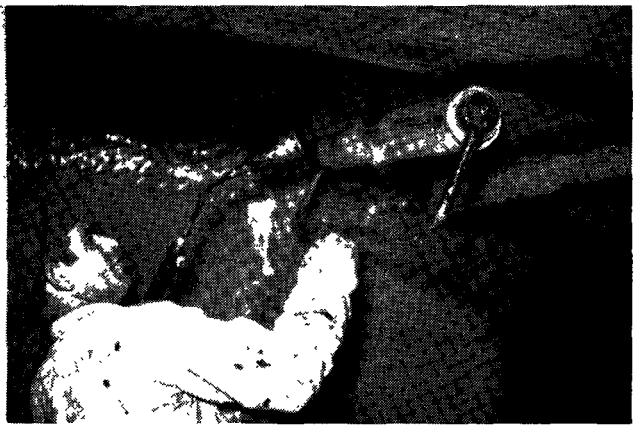


Figure 5. Flame application of bituminous coating to cavern vault

The experimental cavern is located some 50 m from the axis of the main reactor cavern, where geologic and hydrologic conditions are very similar (Fig. 1). The experimental cavern with a volume of 140 m^3 and a surface of 170 m^2 is closed by two concrete walls and two pressure resistant steel doors forming an entrance air-lock. A pipe penetration equipped with a motor valve (V_3) can connect the volume of the experimental cavern to its drainage concrete and the surrounding rock mass. Other air-tight penetrations provide adequate cable and piping connections to the exterior. In order to investigate the gas flow into the adjoining rock, various steel probes (S_1 to S_{16}) have been inbedded into the rock at depths varying from 75 to 427.5 cm (Fig. 3).

With the help of an array of mercury manometers arranged on a pannel and photographed at various time intervals, the evolution of the pressure in the cavern, in the air-lock, and in the various probes could be recorded simultaneously and with sufficient accuracy (Fig. 4). Plastic tubes passing directly through the steel doors connected the various probes to the corresponding manometers. Compensating air input was measured by a "Metric" type flow-meter with an accuracy of better than 4%. During the tests, a thermo-hygrograph was placed in the cavern.

The types of lining successively tested in the experimental cavern are described as follows:

Lining 0 :

Bare rock, 30 cm reinforced concrete for the two air-lock walls.

Lining 1 :

20 cm of porous concrete cast on the floor and side walls, vault of the cavern cast with 20 cm of reinforced concrete, all walls covered with 2 cm of grout.

Lining 2 :

All walls hot-lined with two overlapping coatings of bitumen, each containing a corrugated aluminium foil, of 0.1 mm thickness; all surface then covered with 2 cm of filling grout (Figs. 3 and 5).

Lining 3 :

30 cm of reinforced concrete on all surfaces.

Lining 4 :

Two coatings of epoxy resin applied to all surfaces.

In order to achieve a satisfactory leak-tightness of the cavern, the sandwiched bituminous coating was extended to overlap the steel frames of the air-lock doors and penetrations, where it was located by bolting the corresponding flanges (Fig. 6).

Leak-tightness of the experimental cavern has been subsequently investigated for the various types of

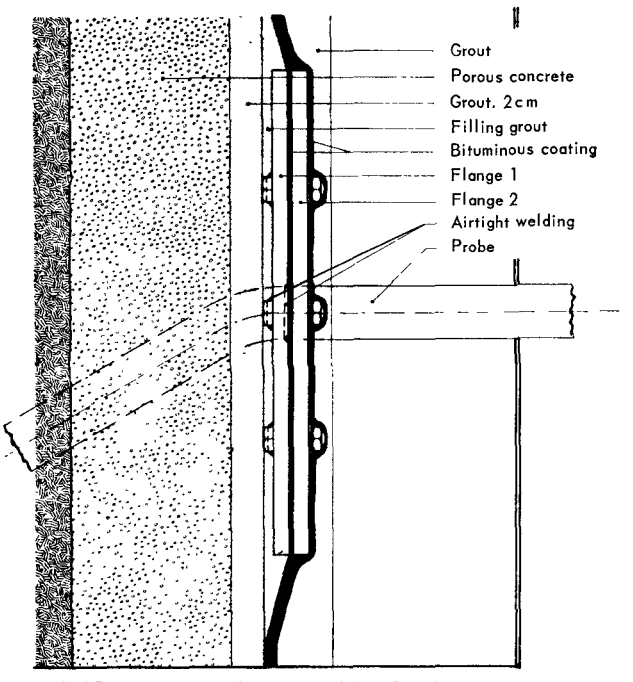


Figure 6. Airtight sealing of pipe penetration

lining mentioned above. Of the two methods applied, the first consists in establishing a rapid overpressure in the cavern and recording the following pressure decay. The second consists in maintaining a constant overpressure in the cavern by means of compensating losses through variable air input. The first method is particularly simple since pressure decay is exponential so that a time of relaxation can be attributed to each type of lining. These vary from 400 minutes for the bare cavern to some 13 000 minutes corresponding to the bitumen lining of type 2. For an initial overpressure of one atmosphere these values correspond to a leak-rate of 20 m^3/h for the bare cavern and a leakage of 0.4 m^3/h for the lining of type 2. The second method was applied only for the barec avern (lining 0), since for very long relaxation times it is a delicate operation to adjust the air input so as to keep the pressure in the cavern constant. Pressure equilibrium was established at five different pressure levels (Fig. 7), yielding five discrete points on the continuous curves obtained with the first method of pressure decay. From these it became apparent that already the bare rock was relatively airtight which would confirm its homogeneity and the absence of large fissures. Applications of the bituminous coating (lining 2) considerably increased the leak-tightness of the cavern which subsequently could not be improved even by the surface application of epoxy resin.

Gas progression in the rock due to flowing and subsequent molecular diffusion

Gas flow in the rock has been investigated by recording the evolution of pressure in the various rock probes. Especially the tests conducted with lining 2 show typical results as shown in Fig. 8. At the initial overpressure of 485 torr, the relief decompression valve was remotely opened thus connecting the experimental cavern with the surrounding rock. After a very short time lapse of some 26 seconds, the air pressure distributed uniformly throughout the drainage system as recorded by the probes G_1, G_2 and G_3 (cf. Figs. 3 and 8). The following pressure decay is exponential. From the curves of pressure response in the various probes it becomes apparent that the pressure build-up is slower with increasing depth, as shown clearly in the curves for probes S_9 to S_{13}. From these measurements the pressure distribution as a function of depth in the rock could be assessed for various time intervals as shown in Fig. 9. For the first minutes, the curves correspond to the solution of the diffusion equation for a plain boundary condition and an initial pressure step. Introducing experimental values for rock permeability and porosity, as well as air viscosity and a reference value for the pressure into the term for the diffusion coefficient, the recorded pressure distribution agrees well with theoretical values. For longer times, however, the effect of pressure decay in the cavern cannot be taken into consideration within a frame of a simple mathematical model. The envelope of these curves show that the gas flow will not affect the equilibrium pressure in the rock beyond a maximum distance of some 7.5 m from the cavern. Similarly, by balancing the air masses initially contained in the experimental cavern and in the porosity of the rock, one can show that gas progression is actually limited to the perturbed zone mentioned above. In Fig. 9 the anomalies registered in the pressure distribution around probes S_9 and S_{11} could not be fully explained. From these tests, one can conclude that gas release from a pressurized cavern

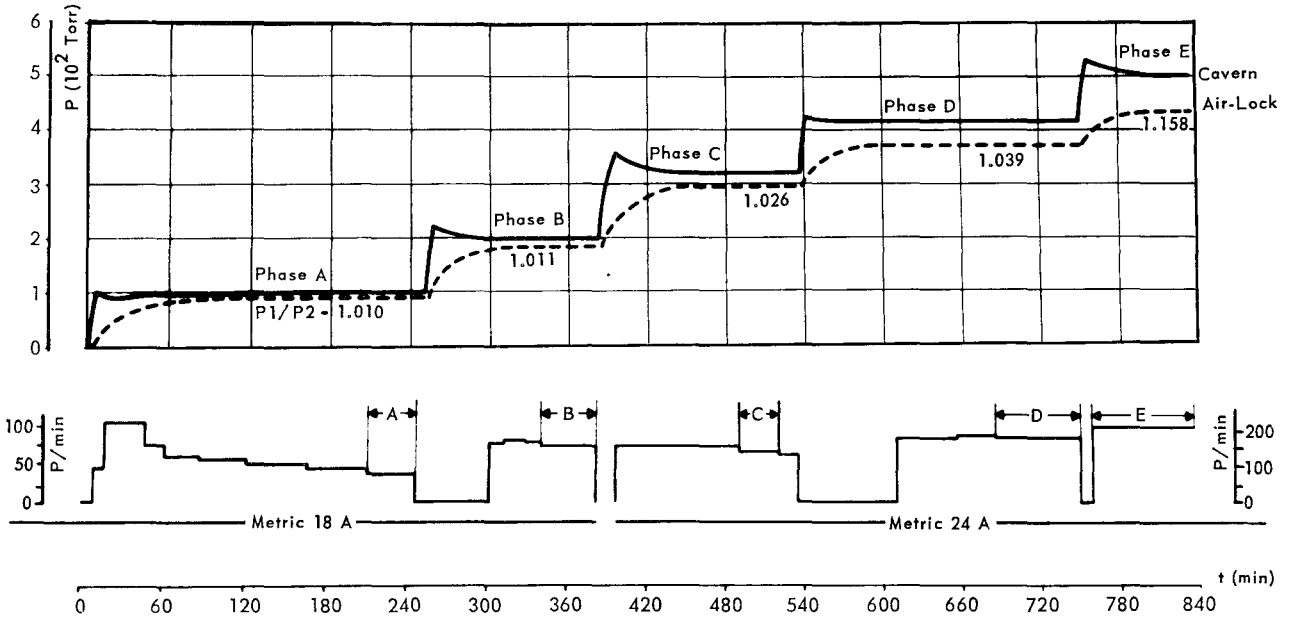


Figure 7. Cavern leakage at constant pressure determined by air-input compensation

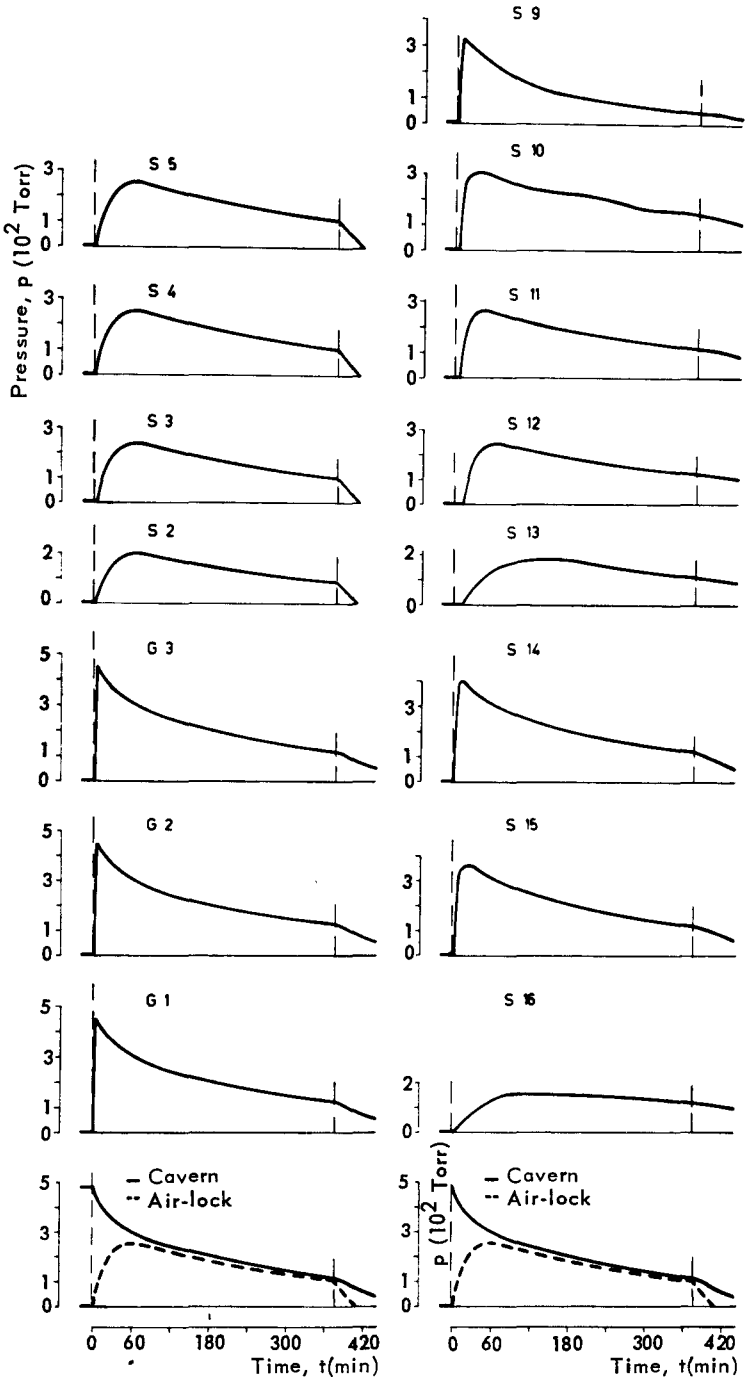


Figure 8. Evolution of pressure in the rock as recorded in probes during gas release to drainage system opening valve V3 (Lining type 2)

to the surrounding porous rock is possible, that it occurs uniformly and affects only the immediate surrounding of the cavern. Gas released to the rock will drive back stagnant air originally contained in the porosity of the rock and replace it without mixing.

After initial gas flow in the rock has stopped, further progression of gases occurs solely through molecular diffusion governed by the corresponding gradient of concentrations and by the diffusion coefficient which depends on the properties of the rock and the gas under consideration [1, 2]. The theoretical interpretation of this phenomenon was checked experimentally at Lucens by introducing Freon-12 (C Cl_2 F_2) in one of the rock probes at a depth of 30 cm at zero overpressure. A freon-detector placed at the surface of the rock recorded the first signal after 254 minutes. Knowing the initial concentration of freon in air and the sensitivity of the detector, the diffusion coefficient

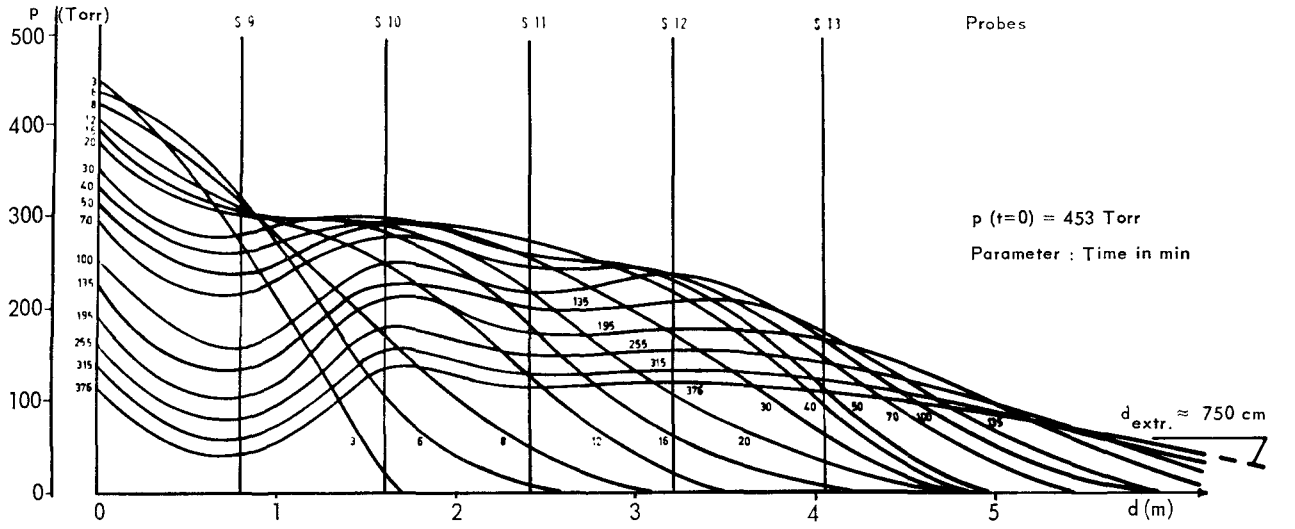


Figure 9. Evolution of pressure distribution during gas release to the rock

for freon could be estimated. Within the limits of accuracy, the measured value corresponds to the theoretical one of 0.9×10^{-3} cm²/s. On the other hand, injecting freon into a rock probe under similar conditions but with an overpressure of 3 atmospheres, the first freon signal occurred after some six seconds. These tests confirm that after pressure has decayed, subsequent progression by molecular diffusion is very slow, thus allowing for extended gas retention in the rock.

CONCLUSIONS

Experimental evidence shows that site conditions at Lucens are in all respects favourable for an underground containment in porous rock. This type of containment appears to be particularly applicable to gas cooled reactors for which, following a severe reactor accident, the overpressure in the containment cannot be fully reduced only by cooling effects and condensation. Rapid reduction of post-accident overpressure in the containment certainly constitutes the most effective means to reduce leakage. Due to the considerable retention of fission products in the rock mass, specification for containment leak-tightness will be less severe than for a corresponding above ground containment. Accordingly, construction techniques for such cavern containment can be simple and conventional, similar to the ones used for tunnels. Furthermore, underground construction will considerably increase resistance to overpressure and will allow problems of external radiation protection to be disregarded. Special care, however, must be given to the layout and construction of cavern accesses. Even for these elements, specifications for leak-tightness will be less severe in as much as all leakage from the main reactor cavern, which is not channelled towards the rock mass, will invariably be collected in the adjacent caverns where, due to the internal ventilation system, airborne contaminants will be extensively removed by means of fine filters and iodine filters, before exhausting offgases to the atmosphere through the high ventilation stack.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/459 Suisse

L'enceinte de confinement souterraine de la centrale nucléaire expérimentale de Lucens

par E. Binggeli *et al.*

La centrale nucléaire expérimentale actuellement en construction à Lucens aura une puissance de 30 MW(th) et sera équipée d'un réacteur à tubes de force modéré à l'eau lourde et refroidi au gaz carbonique.

Le choix de la construction souterraine ayant été fait, on a cherché à établir une enceinte de confinement économique qui, tout en utilisant des techniques aussi simples et conventionnelles que possible, mette à profit les propriétés naturelles du site pour contribuer à assurer la protection de ses environs contre les effets dangereux de l'accident maximal hypothétique.

Des études approfondies des conditions du site, portant principalement sur la géologie et l'hydrologie, ont montré la possibilité d'utiliser le rocher poreux (grès molassique sédimentaire) après l'accident, d'une part comme filtre capable de fixer les particules radioactives et d'autre part comme lieu de stockage apte à retenir les gaz radioactifs pendant un temps suffisamment long pour qu'ils ne puissent exercer d'effets nocifs aux environs de la centrale.

De nombreux essais effectués en laboratoire et sur place, et en particulier ceux faits dans une caverne expérimentale creusée sur le site, ont permis d'étudier les différents mécanismes intervenant pendant l'évacuation des gaz dans le rocher et de mettre au point les différentes techniques à utiliser pour la construction de la caverne du réacteur.

Ainsi, la protection en cas d'accident grave du réacteur est assurée d'abord par la caverne fermée dont les parois sont pourvues d'un revêtement supplémentaire en multicouche « bitume-aluminium » noyé dans l'épaisseur du béton. La surpression qui s'établit dans l'enceinte de confinement lors d'un tel accident, et dont une fraction importante subsiste même après refroidissement de l'atmosphère et condensation de la vapeur d'eau, peut être abaissée en ouvrant des vannes, équipées de filtres fins et disposées dans les parois de la caverne de façon à permettre aux gaz qu'elle contient (CO_2, air, gaz rares et vapeurs d'halogènes) de pénétrer dans les pores du rocher. Le risque représenté par cette atmosphère en surpression et fortement contaminée se trouve ainsi supprimé.

Les essais effectués dans la caverne expérimentale ont montré que l'écoulement des gaz dans le rocher reste limité au voisinage immédiat de la caverne et que la progression de ces gaz au-delà de la zone d'écoulement a lieu très lentement par diffusion moléculaire.

Enfin, le mémoire rappelle les exigences imposées à l'enceinte de confinement et montre, en se basant sur des résultats d'essais, qu'elles sont effectivement satisfaites par les conditions naturelles du site alliées aux caractéristiques de la construction.

A/459 Швейцария

Защитная оболочка подземного реактора опытной атомной электростанции в Люсенсе

Э. Бинджели *et al.*

Опытная атомная электростанция, строящаяся в настоящее время в Люсенсе (Швейцария), тепловой мощностью 30 *Мвт* будет оснащена атомным реактором с трубами под давлением, в котором в качестве замедлителя будет использована тяжелая вода, а в качестве теплоносителя — углекислый газ.

После того как был выбран вариант строительства подземного реактора, начались разработки экономичной защитной оболочки, при создании которой использовались бы не только по возможности простые и доступные технические средства, но и природные условия местности для того, чтобы обеспечить защиту окружающих окрестностей от опасных последствий непредвиденного облучения, которое только можно представить.

Тщательные изучения условий местности, главным образом геологические и гидрологические, указали на возможность использовать пористую горную породу (мягкий осадочный песчаник), которая в случае аварии может быть, с одной стороны, использована как фильтр, способный задерживать радиоактивные частицы, а с другой стороны, как место скопления радиоактивных газов, способное удерживать эти газы в течение времени, достаточного для того, чтобы они уже не могли оказывать вредного влияния на местность, окружающую электростанцию.

Многочисленные испытания, проведенные в лабораториях и непосредственно на месте, в частности испытания, проведенные в экспериментальной пещере, вырытой в скальном грунте, позволили изучить различные явления, происходящие во время выхода газа в скальные породы, и разработать различные технические средства, которые могут быть использованы при сооружении котлована для атомного реактора.

Так, например, в случае серьезной аварии реактора защита обеспечивается прежде всего самой закрытой пещерой, стенки которой дополнительно покрыты несколькими слоями битума с алюминием, залитыми, в свою очередь, толстым слоем бетона. Избыточное давление, которое создается внутри защитной оболочки во время аварии, значительная часть которого все еще остается даже после охлаждения атмосферы и конденсации водяного пара, можно снизить, открыв вентили, оборудованные тонкими фильтрами и расположенные в стенках пещеры с таким расчетом, чтобы позволить находящемуся в ней газу (углекислый газ, воздух, инертный газ и галоидные пары) проникать в поры скальных пород. Таким образом можно устранить опасность, которую представляет эта сильно зараженная атмосфера, находящаяся под избыточным давлением.

Испытания, проводившиеся в экспериментальной пещере, показали, что утечка газа в скальные породы ограничивается геологической средой, непосредственно окружающей экспериментальную пещеру, и что проникновение этих газов за пределы зоны его утечки происходит очень медленно в силу молекулярной диффузии газов.

Наконец, в докладе описываются требования, которые предъявляются к защитной оболочке

подземного реактора, и на основании результатов испытаний делаются выводы, что эти требования могут в значительной степени быть удовлетворены за счет природных условий местности, отвечающих характеру строительства.

A/459 Suiza

El recinto de contención subterráneo de la central nuclear experimental de Lucens

por E. Binggeli *et al.*

La central nuclear experimental que se está construyendo actualmente en Lucens tendrá una potencia de 30 MW(t) y estará equipada con un reactor de tubos de presión, moderado por agua pesada y refrigerado por dióxido de carbono.

Una vez elegida la localización subterránea, se ha tratado de conseguir un recinto de contención económico utilizando al máximo las técnicas más sencillas y tradicionales posibles y aprovechando las propiedades naturales del emplazamiento para que contribuyan a asegurar la protección de los alrededores contra los efectos peligrosos del accidente máximo hipotético.

Los estudios detallados de las condiciones del terreno, especialmente sobre la geología e hidrología, han demostrado la posibilidad de utilizar la roca porosa (arenisca arcillosa sedimentaria) después del accidente, por una parte, como filtro capaz de fijar las partículas radiactivas y, por otra, como lugar de almacenamiento, capaz de retener los gases radiactivos durante un tiempo suficientemente largo, para que no puedan ejercer efectos nocivos en los alrededores de la central.

Numerosos ensayos efectuados en el laboratorio y en el terreno, en particular los realizados en una caverna experimental perforada en el emplazamiento, han permitido estudiar los diferentes mecanismos que intervienen en la evacuación del gas en la roca y poner a punto las diversas técnicas a aplicar en la construcción de la caverna del reactor.

De este modo, la protección en caso de accidente grave del reactor está asegurada, primeramente por la caverna cerrada, cuyas paredes tienen un revestimiento suplementario de varias capas de « Bitume-aluminium » embebido en el espesor de hormigón. La sobrepresión que se produce en el recinto de contención después de tal accidente, de la cual subsiste una fracción importante aún después del enfriamiento de la atmósfera y la condensación del vapor de agua, puede reducirse abriendo válvulas, equipadas con filtros finos, colocadas en las paredes de la caverna, de forma que los gases que contiene (CO_2, aire, gases nobles y vapores de halógenos) puedan penetrar en los poros de la roca. Se elimina de esta forma el riesgo que representa esta atmósfera fuertement contaminada y a sobrepresión.

Los ensayos efectuados en la caverna experimental han demostrado que la penetración del gas en la roca queda limitada a las proximidades de la caverna y que la progresión de los gases fuera de esta zona se produce muy lentamente por difusión molecular.

Por último, la presente memoria recuerda las exigencias impuestas al recinto de contención y demuestra, basándose en los resultados de los ensayos, que quedan satisfechas adecuadamente por las condiciones naturales del emplazamiento combinadas con las características de la construcción.

Aspectos de seguridad de los reactores del tipo DON

por A. Alonso * y B. L. Hoffman **

El DON es un tipo de reactor de uranio natural, refrigerado por un líquido orgánico y moderado por agua pesada. La JEN, desde hace ya varios años, estudia diversos aspectos de este tipo de reactores, con el fin de construir en el futuro centrales núcleo-eléctricas grandes.

La JEN ha concedido también gran importancia a los aspectos de seguridad de estos reactores, llevándose a cabo trabajos teóricos y experimentales en este sentido. En la presente memoria se resumen y generalizan las conclusiones más importantes de dichos trabajos. Se analizan también distintos tipos de cubiertas y de sistemas de contención haciendo uso de un procedimiento analítico desarrollado para el prototipo de reactor DON de la JEN, pero de aplicación muy general.

CARACTERÍSTICAS GENERALES DE LOS REACTORES DEL TIPO DON Y DE LOS REFRIGERANTES ORGÁNICOS

Descripción

El utilizar compuestos orgánicos como refrigerantes de reactores simplifica y abarata los materiales de construcción y el diseño de los componentes. Sin embargo, el que el moderador sea un fluido distinto introduce muchas complicaciones en el diseño del conjunto. El hecho de que la temperatura del moderador tenga que ser baja, a pesar de estar cerca del líquido orgánico, cuya temperatura es de unos 345 °C, lleva consigo el utilizar barreras térmicas efectivas y seguras entre el canal del refrigerante y el tanque del moderador. Se pueden cumplir estos requisitos de muchas formas; en el caso del reactor DON de la JEN, se ha seleccionado un sistema de calandria. Dicho tanque contiene al moderador y está atravesado por muchos tubos de presión, que encierran el combustible y los canales de refrigeración, y que forman un anillo con la pared exterior del buto de la calandria; anillo, que al llenarlo de gas, constituye una barrera térmica.

El sistema de distribución del refrigerante puede ser de varias formas; se pueden utilizar dos tanques de recogida situados a la entrada y a la salida del núcleo; se ha propuesto también un solo tanque dividido en dos partes, si se emplean elementos combustibles del tipo bayoneta; también se han considerado tuberías flexibles que unan cada canal con sendos toros de entrada y salida del refrigerante. Estudios recientes [1] indican que el último sistema es el más práctico. En otra memoria presentada en esta Conferencia [2] de discuten los detalles de dicho proyecto y las razones de la elección del tercer método para el reactor DON de la JEN.

La selección de dicho sistema afecta a la seguridad del reactor, a causa de que el combustible puede quedar al descubierto al romperse, en determinadas condiciones, las tuberías de entrada del refrigerante. Por ello, hay que prever sistemas especiales que garanticen la refrigeración del combustible en estas condiciones. En el caso del reactor DON de la JEN, dicho sistema especial consta de un sumidero en la cavidad inferior, de un tanque de recogida y de una bomba automática de circulación.

El compuesto orgánico propuesto para el reactor DON de la JEN es una mezcla comercial de terfenilos, cuyas propiedades físico-químicas se han estudiado en la JEN [3] y en otras instituciones [4]. Estos líquidos orgánicos poseen varias ventajas peculiares, que no reunen los demás refrigerantes.

Una de las características más interesantes de dichos líquidos orgánicos, que fue probablemente una de las que influyeron más en la selección de estos materiales como refrigerantes de reactores, es la de poder utilizar materiales normales en el sistema de contención, lo que en definitiva reduce los gastos de capital. La tensión de vapor de dichos líquidos es baja, lo que permite funcionar a temperaturas relativamente elevadas, que se traducen en una mayor eficacia del ciclo de vapor sin necesidad de elevar la presión del sistema. La baja tensión de vapor es causa también de que sea relativamente pequeña la energía liberada en el caso de roturas o fisuras en las tuberías, lo que permite prescindir de los sistemas de contención absolutos, que se utilizan para limitar las consecuencias de los accidentes nucleares.

Los líquidos orgánicos utilizados como refrigerantes no reaccionan ni con los materiales de la estructura y de las vainas ni con las aleaciones o compuestos de

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uranio que se utilizan normalmente como combustibles. Esta compatibilidad química limita las consecuencias de la rotura de las vainas, ya que en este caso la radiactividad que pasa del combustible al refrigerante es muy pequeña, además se puede explotar el reactor con seguridad aun teniendo muchas vainas averiadas ye stando una parte importante de la superficie del combustible expuesta al refrigerante. La corrosión de los materiales es muy pequeña y el líquido refrigerante puede mantenerse puro, por lo que la actividad contenida en aquel es, por lo general, muy pequeña. Por otra parte, la actividad inducida en el propio refrigerante es escasísima. Estas características reducen las consecuencias de los escapes del refrigerante. En el caso de que esto suceda, solamente pasan a la atmósfera parte de los gases de fisión, y para que escapen los otros productos de fisión el líquido orgánico ha de evaporarse o arder, lo que se puede evitar fácilmente. Por tanto, es de esperar que en dichos accidentes la actividad en el edificio sea muy pequeña, lo que permite simplificar las especificaciones sobre el sistema de contención, con el ahorro consiguiente.

De todos los materiales combustibles que se pueden utilizar en este tipo de reactores, el más atractivo por sus propiedades mecánicas, físicas y nucleares es el carburo de uranio. Aunque todavía haya que investigar algunos aspectos, se cree que la tecnología de este compuesto se halla desarrollada como para proponerlo en un reactor prototipo [1].

La física del núcleo en relación con la seguridad

La temperatura de los neutrones en los reactores del tipo DON es baja y el espectro correspondiente se encuentra también desplazado hacia las energías pequeñas, a causa de la temperatura del moderador. Este hechos afecta a los aspectos cinéticos del reactor, porque el coeficiente de temperatura del moderador, aun siendo negativo, experimenta una cierta tendencia hacia los valores positivos. La vida media de los neutrones es, por otra parte, relativamente larga en los reactores moderados por agua pesada, lo que se traduce en períodos también largos para una inserción dada de reactividad. A este respecto el reactor es más fácil de controlar.

Una característica peculiar de los reactores moderados por D_2O y por Be es la producción de foto-neutrones. El efecto de los foto-neutrones es como si aumentase la fracción de los neutrones retardados, lo que originaría valores más elevados de la potencia en el caso de que el reactor alcanzase el estado crítico solo con los neutrones inmediatos.

El parámetro más intimamente ligado a la seguridad del reactor es el coeficiente de potencia. De entre los parámetros que intervienen en la determinación de dicho número hay que destacar los coeficientes de temperatura del combustible, del moderador y del refrigerante. El coeficiente de temperatura del combustible es el que, por lo general, ejerce un efecto inmediato sobre los estados transitorios y es normalmente negativo a causa del efecto Doppler. El coeficiente de temperatura del refrigerante en el caso del reactor DON de la JEN es positivo, ya que absorbe muchos más neutrones que el moderador y, por otra parte, es un veneno nuclear poco activo en el conjunto del sistema. El coeficiente de temperatura del moderador es, por lo general, negativo aunque próximo a cero, a causa del efecto de la temperatura de los neutrones mencionado antes. Al progresar el quemado del combustible, dicho coeficiente tiende a ser nulo o en algunos casos positivo.

Estabilidad

Del valor del coeficiente de potencia no se puede deducir mas que una idea preliminar de la estabilidad dinámica del sistema, ya que las relaciones de tiempo entre los distintos parámetros que contribuyen al coeficiente de potencia son las que realmente definen las características dinámicas del reactor.

Se ha estudiado la estabilidad dinámica de este tipo de reactores utilizando máquinas analógicas [1]. En general, ni se han obtenido oscilaciones divergentes ni siquera sostenidas, aun con aumentos instantáneos de reactividad de hasta 25 céntimos. Esta conducta se atribuye principalmente al coeficiente de temperatura del combustible, que actúa antes que ningún otro a causa de la elevada capacidad calorífica del combustible y al retraso que existe en el paso de calor del combustible al refrigerante. El ánulo de gas que existe entre el combustible y la vaina, en el caso de combustibles cerámicos, amortigua con efectividad los estados transitorios, porque introduce un retraso muy grande en el aumento de temperatura del refrigerante comparado con la rapidez con que aumenta la temperatura del combustible.

Se deduce, por tanto, que el tipo de reactor analizado se puede diseñar de modo que posea características inherentes de estabilidad y que se puede controlar y proteger utilizando instrumentos y sistemas convencionales.

REVISIÓN DE LOS ACCIDENTES VEROSÍMILES

Por lo general, para que un accidente nuclear afecte a terceros o sea un riesgo para la salud y la seguridad del público, han de escapar a la biosfera, de forma incontrolada, cantidades importantes de productos de fisión. Aunque los accidentes de esta naturaleza sean verosímiles, la probabilidad de que tengan lugar es muy pequeña, ya que para que ocurran ha de haber varias averías simultáneas, tanto en el sistema de protección del reactor como en las barreras físicas

incorporados al efecto. Se discuten seguidamente algunos accidentes de esta clase que pueden tener lugar en los reactores del tipo DON, así como las medidas necesarias para contrarrestar sus efectos.

Para que los productos de fisión contenidos en el combustible salgan al exterior hay que romper tres barreras distintas; a saber: la vaina del elemento combustible, las tuberías u otros componentes del sistema primario y el sistema de contención propiamente dicho. Por otra parte, el coeficiente de potencia de este tipo de reactores es negativo; por tanto, el construir sistemas de control y de protección adecuados es tarea fácil. De aquí se deduce que, si los sistemas de protección y de seguridad funcionan normalmente, no serán posibles los accidentes que ocasionen el escape incontrolado de los productos de fisión a la biosfera. Para que haya tal escape, hay que considerar accidentes no protegidos y, además, averías múltiples y simultáneas.

Accidentes en el manejo de los elementos combustibles

Las operaciones de reciclado llevan consigo el manejo de elementos combustibles muy radiactivos. Por muy perfecto que sea el dispositivo de carga, existe siempre la posibilidad de que se caiga al suelo todo o parte del elemento combustible que se maneje. Para evitar errores que pueden dar lugar a accidentes, se dispone de enclavamientos de seguridad, que impiden el comenzar una operación de reciclado sin haber completado la anterior con éxito. Si los dispositivos de reciclado se proyectan bien, entonces tendrán que ocurrir varias averías simultáneas para que tenga lugar el accidente.

Fusión del núcleo

Parte del núcleo del reactor podría llegar a fundir si se añadiesen súbitamente cantidades importantes de reactividad y no estuviese protegido el estado transitorio resultante, o si se bloquease el refrigerante. Sin embargo, las características propias del núcleo limitarían el porcentaje del núcleo que fundiría.

Para que la actividad así liberada pueda llegar a contaminar la biosfera, tienen que existir otras dos averías, una en el sistema primario de refrigeración y otra en el sistema de contención, si existe. Se ha sujerido utilizar sistemas de contención absolutos para evitar de esta forma el escape de los productos de fisión; sin embargo, este procedimiento es muy caro y, por lo general, no se considera necesario. Un edificio normal es a veces suficiente, si se tienen en cuenta ciertos detalles tales como la construcción de piscinas de condensación y de equipos de descontaminación, tanques de retención, etc.

Combustión del orgánico

Todas las instalaciones que contengan hidrocarburos corren los riesgos de incendio y de explosión; si se cumplen ciertas condiciones. Con un proyecto adecuado se puede evitar la presencia de dichas condiciones; además, aun en el caso de que se produjesen, se dispone de distintos tipos de extintores que, al menos, limitan el daño.

Para que un accidente de este tipo tenga consecuencias radiológicas importantes, es necesario que el refrigerante se halle contaminado, lo que puede suceder si antes ha fundido parte del combustible. Además, para que el refrigerante se escape de su sistema es necesario que éste se haya roto. Si existen las condiciones para la ignición y falla el sistema contraincendios, entonces habrá un fuego, de lo que puede resultar la dispersión de la radiactividad. Se limita la cantidad de actividad puesta en libertad evitando la llegada del aire al lugar del fuego y tratando de descontaminar los gases de la combustión antes de que escapen a la biosfera.

Se puede decir, por tanto, que los accidentes que ocasionen escapes incontrolados e importantes de radiactividad solo pueden tener lugar en este tipo de centrales nucleares en condiciones muy poco probables. Además, los métodos de protección usados normalmente en otros tipos de centrales se pueden utilizar también en las del tipo DON, con la posible ventaja sobre aquellas que las propiedades de los líquidos orgánicos utilizados como refrigerantes permiten obtener mayor rendimiento de dichos sistemas de protección.

ANÁLISIS DE VARIOS SISTEMAS DE CONTENCIÓN

Se describen en este apartado varios sistemas de contención de coste y complejidad variables y se analiza cada uno de ellos en relación con una central nuclear del tipo DON.

El proyectista debe resolver el problema de la contención de la central nuclear después de considerar con cuidado los accidentes verosímiles, las características peculiares a la central en estudio y, naturalmente, ha de tener en cuenta la seguridad del público. La decisión final debe basarse en los aspectos económicos, siempre que sean compatibles con las consideraciones anteriores. El instalar dispositivos de contención de eficacia superior a la necesaria se traducirá en una seguridad mayor pero, por lo general, aumentará innecesariamente el coste de la instalación.

Se han analizado ocho sistemas distintos de contención de coste y complejidad crecientes. El componente fundamental de los cuatro primeros sistemas de contención es un edificio cuya velocidad de fugas en las condiciones producidas durante el accidente máximo

verosímil, que se supone resulta en la fusión de 1/3 del núcleo, varía desde 100% hasta 0,1% por día en saltos logarítmicos para mostrar el efecto de este parámetro al pasar de edificios convencionales a estructuras más herméticas.

En el sistema V se demuestra el efecto de una chimenea de 50 m de altura y de un sistema de ventilación de tipo industrial, cuyos filtros tienen una eficacia de 95% para retener sólidos, de 90% para retener halógenos y teluro y nula para retener los gases nobles. En el sistema VI la eficacia de los filtros es de 99,9% para sólidos, de 99% para halógenos y teluro y también nula para retener los gases nobles. Tales filtros se pueden diseñar y construir y el ejemplo sirve para mostrar el efecto de la eficacia del filtro. En ambos casos se supone que los ventiladores cambien el aire del edificio dos veces por hora.

En el sistema VII se supone que los productos de fisión borbotean a través de una piscina de retención, que en realidad equivale a un filtro de eficacia media; los productos de fisión que no condensan escapan a la atmósfera a través de la chimenea. Se ha estudiado también en este caso el efecto de la velocidad de fusión del núcleo. En el caso VII-a se supone que el núcleo se funde instantáneamente y que los gases radiactivos se escapan en muy poco tiempo. En el segundo caso, VII-b, los procesos de la fusión del núcleo y del escape consiguiente de los productos de fisión duran una hora y, finalmente, en el caso VII-c, el proceso anterior dura unas ocho horas.

El sistema VIII posee, además de la piscina de condensación y entre ella y la chimenea, un sistema de tratamiento de gases, que consta de un tren de columnas de adsorción que retardan el escape de los gases y de un sistema de filtración. Se han estudiado también en este sistema el efecto del retraso introducido en el sistema de tratamiento de gases. En el primer caso la duración total del escape es de unas ocho horas y en el segundo de cuatro días.

El análisis de dichos sistemas se ha efectuado en la máquina calculadora digital UNIVAC-UCT de la JEN utilizando el programa NUBE *. Dicho programa es muy flexible y permite tener en cuenta: la precipitación y condensación de los vapores y aerosoles sobre el suelo y paredes del edificio, así como sobre el terreno; la condición de difusión atmosférica, habiéndose elegido la condición *F* de Pasquill con vientos constantes de 2 m/s; y el tiempo y modo de recibir las dosis, habiéndose seleccionado las condiciones que definen el radio de exclusión tal y como se emplea en las normas vigentes en los EE.UU. [5]. El programa NUBE analiza las dosis de radiación recibidas por los distintos órganos y por todo el cuerpo. Los resultados obtenidos demuestran que el tiroides es el órgano crítico. Por consiguiente, la dosis normalizada recibida por el tiroides en función de la distancia al reactor se representa en la figura 1 para los cuatro primeros casos, y en la figura 2 para los otros cuatro. Se ha estimado conveniente normalizar la dosis en función del inventario de productos de fisión medido en megavatios·año y del porcentaje que se funde del núcleo. Los resultados se discuten en la sección siguiente.

* El programa NUBE ha sido desarrollado en la JEN para evaluar los emplazamientos de los reactores.

CONCLUSIONES Y APLICACIÓN AL REACTOR DON DE LA JEN

La potencia térmica del reactor DON de la JEN es de 100 MW; fijando en 300 rem la dosis de referencia al tiroides se deduce de la figura 1 que para albergar al reactor en un edificio convencional se necesitaría disponer de una zona de exclusión cuyo radio fuese superior a 2 km, lo que evidentemente es prohibitivo; basta, sin embargo, construir un edificio cuyas fugas durante el accidente máximo sean inferiores a 10% por día para que el radio de exclusión sea inferior a 0,5 km, lo que ya es practicable. Un edificio de dichas características no es difícil de lograr teniendo en cuenta que en los reactores orgánicos la energía potencial almacenada en el refrigerante es pequeña.

De la figura 1 se deduce también que los reactores muy grandes, del orden de los 1 000 MW(t), o se instalan en vasijas de presión o se necesitan sistemas adicionales de filtración o lavado para que sus radios de exclusión sean aceptables. Naturalmente, en el segundo caso hay que garantizar también el funcionamiento adecuado y seguro de dichos sistemas.

El primer fenómeno que se observa en la figura 2 es el efecto de la chimenea, de tal forma que la dosis máxima se recibe a unos 3 km del reactor. En los cuatro primeros sistemas, las dosis disminuyen al aumentar la distancia al reactor, en estos casos dichas dosis controlan el radio de exclusión. En los casos de la figura 2, la zona de baja densidad de población constituye el criterio fundamental.

La adición de un filtro de gran eficacia supone una gran ventaja con respecto al filtro industrial. Sin embargo, para el reactor DON de la JEN sobraría con el primero, reservando el de gran eficacia para reactores de 1 000 MW(t).

Es curioso observar que la adición de una piscina de retención, aunque tenga mayor eficacia para retener los productos de fisión que el filtro industrial, no ofrece ninguna ventaja desde el punto de vista de la seguridad y ello se debe a que en los sistemas V y VI el propio edificio actúa como un sumidero importante de radiactividad, al permitir que los productos de fisión se depositen sobre el suelo y las paredes frías, lo que no ocurre con tanta facilidad en los sistemas VII y VIII Como puede deducirse, las velocidades de fusión de

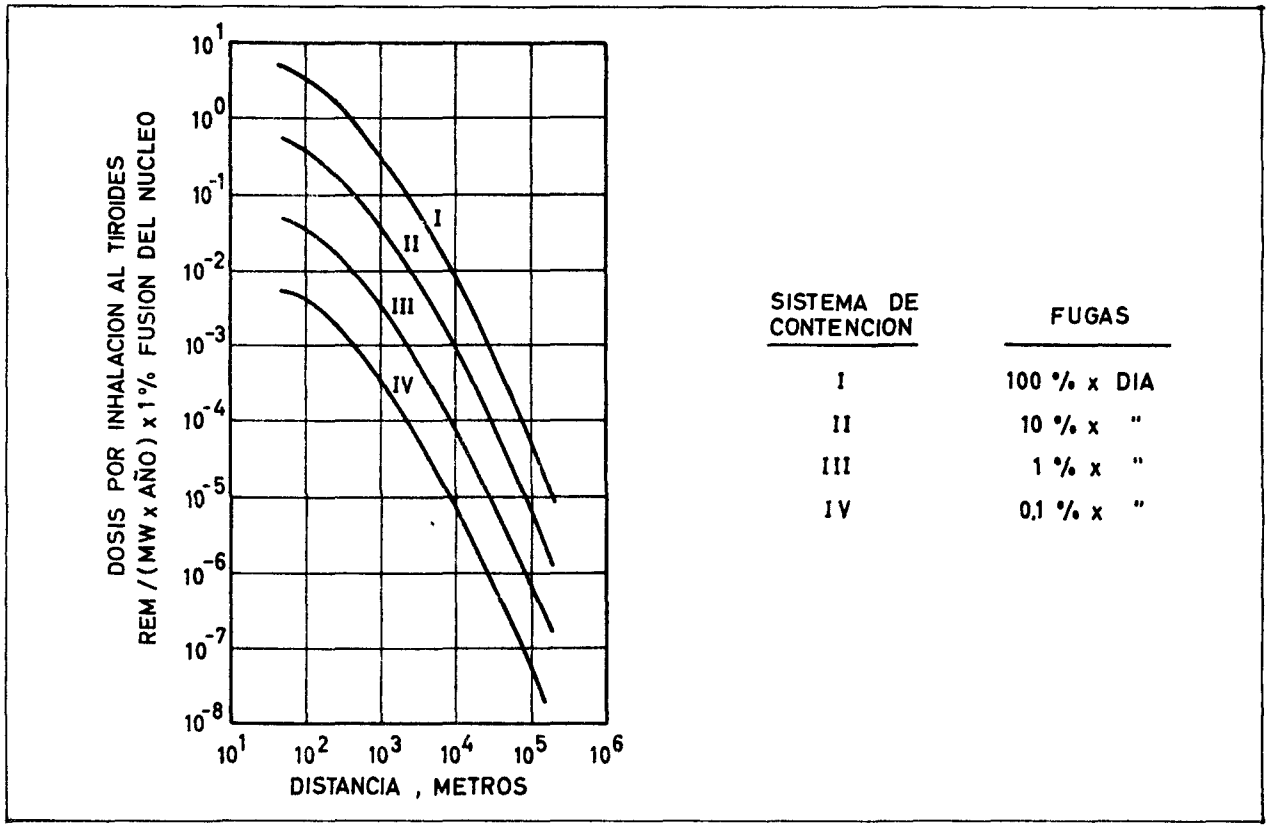


Figura 1. Dosis por inhalación al tiroides en funcion de la distancia al reactor y de las fugas del edificio

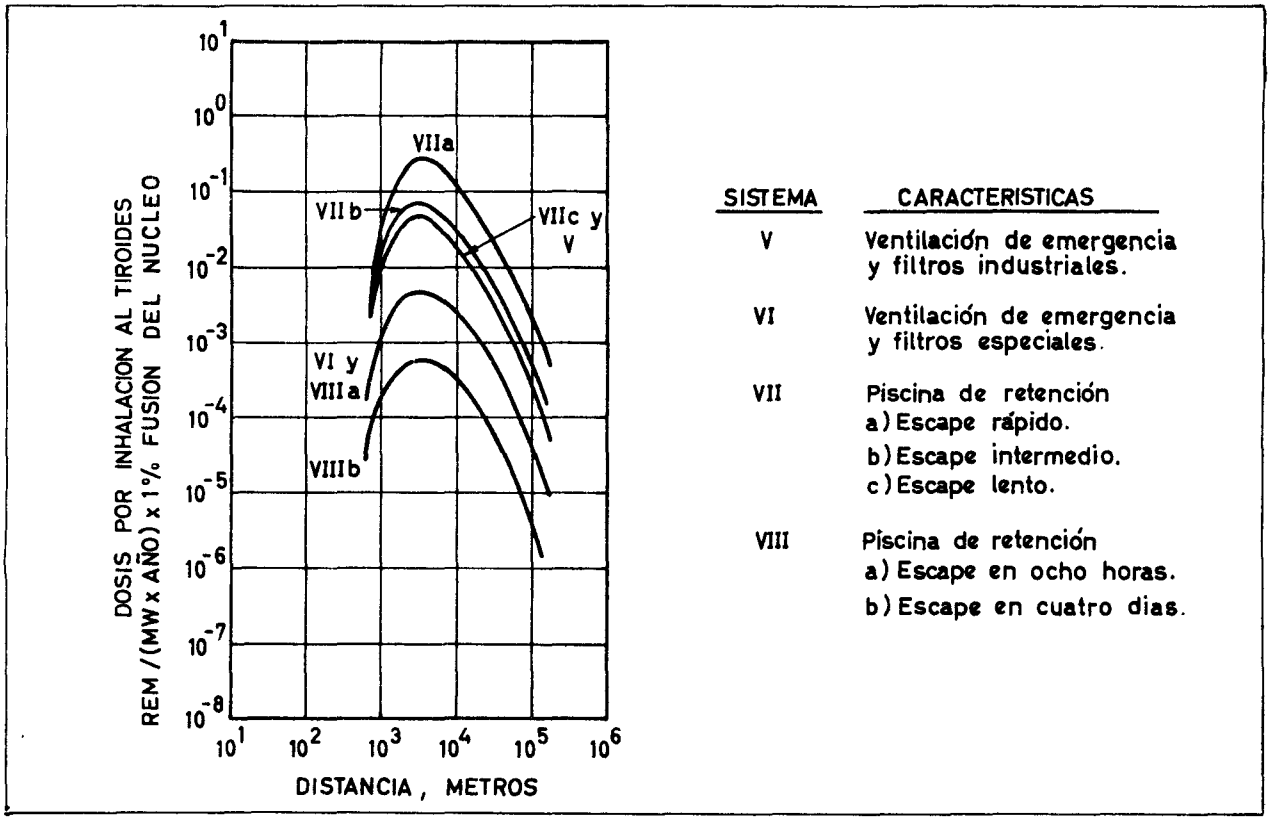


Figura 2. Dosis por inhalación al tiroides en funcion de la distancia al reactor para distintos sistemas de contención

núcleo y del escape de los productos de fisión son también factores importantes. Los casos analizados demuestran que el sistema VII no puede emplearse en el reactor DON de la JEN a no ser que se aumente la eficacia de la piscina de retención o se consiga retrasar el escape de los productos de fisión.

El sistema VIII es el más eficaz de los estudiados. Si el escape tiene lugar en ocho horas, el efecto combinado de la piscina de retención y del filtro industrial es equivalente entonces a un filtro muy eficaz. Si aumenta la duración del escape aumenta considerablemente la efectividad del sistema; como se muestra en la figura 2.

Para el caso del reactor DON de la JEN se ofrecen, por tanto, varias alternativas que han de ser estudiadas desde el punto de vista económico. Se puede construir un edificio con muy pequeña velocidad de fugas, pero sin ser de presión. Cabe también considerar un edificio convencional provisto de chimenea y con un sistema duplicado de ventilación y de filtración de emergencia. Finalmente, sería oportuno el analizar la sustitución del sistema de ventilación por una piscina de retención conectada con la vasija del reactor y de un sistema adecuado de retención temporal de los productos de fisión, siempre que se disponga de medios para impedir que los productos de fisión escapen a la nave del reactor.

Con frecuencia y por motivos es conveniente disponer de ambos sistemas. En este caso, si no se puede definir cuantitativamente qué fracción de los productos de fisión seguirá un camino o el otro, conviene entonces suponer que el 100% de los productos de fisión liberados pueden seguir ambos caminos. Nótese que el uso de una piscina de retención conectada a la cavidad del reactor tiene la ventaja de facilitar las operaciones de descontaminación en caso de accidente. En el caso del reactor DON de la JEN se ha propuesto utilizar ambos sistemas.

EXPRESIONES DE AGRADECIMIENTO

Agradecemos al Grupo de Análisis y Cálculo Numérico de la División de Física Teórica y Cálculo de Reactores la ayuda y colaboración prestadas. En particular, al celo de D. Manuel Domínguez se debe la puesta a punto del programa NUBE utilizado en la segunda parte de esta memoria.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/495 Spain

Safety features of the DON reactor concept

By A. Alonso and B. L. Hoffman

DON is a heavy-water-moderated, organic-cooled, natural-uranium-fuelled type of reactor. A 100 MW(th) version of this concept is being studied at present by the JEN to serve as a prototype for large nuclear power plants.

Safety analyses of this prototype have been published containing theoretical and experimental work and this work is continuing. In these studies, the safety aspects of the whole reactor concept are analysed.

The reactor is a low pressure system due to the physical properties of the organic coolant and the operating conditions of the moderator, and this is one of the major advantages of this reactor concept.

Equations have been developed and programmed to describe the release of radioactive contaminants to the atmosphere from credible accidents. Different types of containment schemes have been analysed to prove that simple engineering features are enough to reduce a release to below the permissible limits.

Mechanical and thermal barriers have to be located within coolant and moderator. The integrity of these barriers has to be maintained throughout the reactor life, thus posing a problem to be studied. A preliminary chemical compatibility study among the various components suggested for the DON prototype shows that no hazardous chemical reactions are likely to occur.

Reactor stability and its influence on nuclear accidents are discussed. The radiological consequences of reactor operation are analysed to prove that reactors of this type can be built and operated in normal locations without undue risk to the health

and safety of the public and without adding costly or novel engineering safety features.

A/495 Espagne

Sûreté des réacteurs du type DON

par A. Alonso et B. L. Hoffman

Il s'agit d'un réacteur à uranium naturel, à eau lourde et à fluide de refroidissement organique. La JEN étudie la construction d'un réacteur de ce type qui aura une puissance de 100 MW(th) et sera le prototype de grandes centrales nucléaires.

On a publié des rapports sur la sûreté de ce prototype, qui contiennent des études théoriques et expémentales. Les travaux se poursuivent. En se fondant sur les conclusions de ces études, les auteurs analysent les problèmes de la sûreté de ce type de réacteur.

L'installation fonctionne à basse pression, du fait des propriétés physiques du fluide organique et des conditions de fonctionnement du ralentisseur. Cette particularité constitue l'un des principaux avantages de la filière.

Les auteurs ont établi les équations décrivant l'échappement dans l'atmosphère des gaz et vapeurs radioactifs libérés au cours d'accidents possibles et les ont programmées pour des calculatrices. Ils analysent différents types de couvercles et de dispositifs de confinement et montrent que, pour maintenir le dégagement de produits toxiques au-dessous des niveaux admissibles, des dispositifs de sécurité très simples suffiront.

Ils ont prévu des boucliers thermiques et mécaniques pour séparer le fluide de refroidissement du ralentisseur. L'intégrité de ces boucliers doit être maintenue pendant toute la période d'utilisation du réacteur; ce problème n'est pas encore résolu. Une étude préliminaire de la compatibilité chimique des différentes matières proposées pour la construction du réacteur DON indique qu'il est tout à fait invraisemblable que des réactions chimiques dangereuses se produisent.

Les auteurs évaluent la stabilité du réacteur et étudient son influence sur les accidents nucléaires. Ils analysent les risques radiologiques mis en jeu par l'exploitation de cette filière et montrent qu'on peut placer ces réacteurs et les exploiter sur des sites normaux sans faire courir un risque inadmissible à la santé et à la sécurité du public et sans avoir à y ajouter des dispositifs de sécurité nouveaux et coûteux.

A/495 Испания

Предохранительные устройства реактора DON

А. Алонсо, Б. Л. Хофман

Установка DON — это реактор на природном уране с тяжеловодным замедлителем и органическим теплоносителем. В настоящее время изучается вариант этой конструкции мощностью 100 *Мвт* (тепл.), который будет служить прототипом реакторов крупных атомных электростанций.

Опубликованный анализ безопасности этого прототипа содержал описание результатов теоретических и экспериментальных работ, посвященных данному вопросу. В настоящее время эта работа продолжается. Она включает полный анализ безопасности реакторов рассматриваемой конструкции.

Реактор является системой низкого давления благодаря физическим свойствам используемых органических веществ и условиям работы замедлителя, что считается одним из основных преимуществ реакторов данного типа.

Были составлены и запрограммированы уравнения, описывающие вероятный выброс радиоактивных загрязнений в атмосферу в результате аварии. Проводится анализ различных типов систем удержания радиоактивных выбросов, показывающий, что простые технические устройства позволяют уменьшить выброс до уровня, не превышающего допустимый.

В теплоносителе и замедлителе необходимо установить механические и тепловые барьеры. Целостность этих барьеров необходимо поддерживать в течение всей кампании реактора, что является проблемой, требующей дальнейшего изучения. Предварительное изучение химической совместимости различных компонентов показывает, что каких-либо опасных химических реакций на прототипе DON, по-видимому, не произойдет.

Обсуждаются вопросы устойчивости реактора и ее влияние на ядерные аварии. Анализируются радиационные последствия работы реактора. Проведенный анализ показывает, что реакторы данного типа без применения дорогостоящих предохранительных устройств могут строиться и работать в любом месте, не представляя особой угрозы для здоровья и безопасности населения.

Factors of economy, safety and hygiene in locating different types of large thermal power stations in Sweden

By S. O. W. Bergström,* and C.-E. Holmquist **

Up to the present, power generation in Sweden has been based on hydro power. Thermal power stations have mainly been designated for peaking and stand-by purposes. Within the next ten to fifteen years, most of the economically usable hydro power will have been developed. Additional generating capacity will therefore be provided in an increasing amount by thermal stations.

When planning for future thermal power developments—using fossil or nuclear fuel—one is in the fortunate situation of starting more or less from scratch, determining the type, size and siting of the plants based on the evaluation of all relevant factors. The present paper tries to establish how the environmental factors influencing the location and design ought to be defined. The study is limited to base load plants and to the period 1965 to 1980.

PRINCIPAL FACTORS OF POWER ECONOMY INFLUENCING PLANNING AND SITING

By principal power economy factors we understand factors that are practically independent of the type of generating machinery to be used and of the particular local conditions at the plant sites. Basically, there are four such principal factors:

(i) Composition of existing power system;
(ii) Growth and geographical distribution of load;
(iii) Limit of unit size and plant capacity;
(iv) Transmission costs.

After defining these factors, an idealized development plan can be made. As all Swedish generating systems are interconnected, jointly-owned stations or stand-by power agreements are common practice. In the following discussion, the Swedish power system will be treated as a whole.

The existing power system

The total generating capacity was 11 200 MW at the end of 1963, of which 8 600 MW was hydro power. Corresponding figures for units produced were 40.5 and 37.8 TWh respectively. The main part of the hydro power is located in the northern part of the country, whereas the power consumption is highest in the southern parts. Figure 1 shows a simplified pattern of the electric power generation and distribution. Figure 2 shows the main high voltage transmission system. Its north-south transmission capacity is at present about 4 000 MW.

Growth and geographical distribution of load

The average annual load increase has for a long time been about 7%. For the future, conservative forecasts [1] figure a slower rate of increase, leading to a total annual generation in 1980 of about 100 TWh. Of this, more than 60% would be generated by hydro plants, the remaining in thermal stations. Figure 3 illustrates this.

For the period 1965 to 1980, one estimates a total additional capacity in *base-load thermal power stations* of about 4 000 MW. It is evident that this thermal power capacity should be located in the southern part of Sweden.

Limit of unit size and plant capacity

A decrease in specific cost with increased unit size seems to be obtainable at least up to 1 000 MW. When doubling the size of a conventional steam unit, the decrease is between 10 and 15% and one estimates that the saving will be even greater for nuclear units. The net saving will, however, be considerably reduced by increased costs for backup capacity. Saving derived from size-independent items (site development, engineering design costs) can, on the other hand, be gained by building multi-unit stations. In that case, the station size will be determined only be otpimization of the whole system, and primarily dependent on the cost of transmission. (The limitations originating from environmental factors are considered in a following section.)

In Sweden, a *unit size* limit of about 300 MW is foreseen for units now in the planning stage, an increase to about double that size may be reasonable for 1980.

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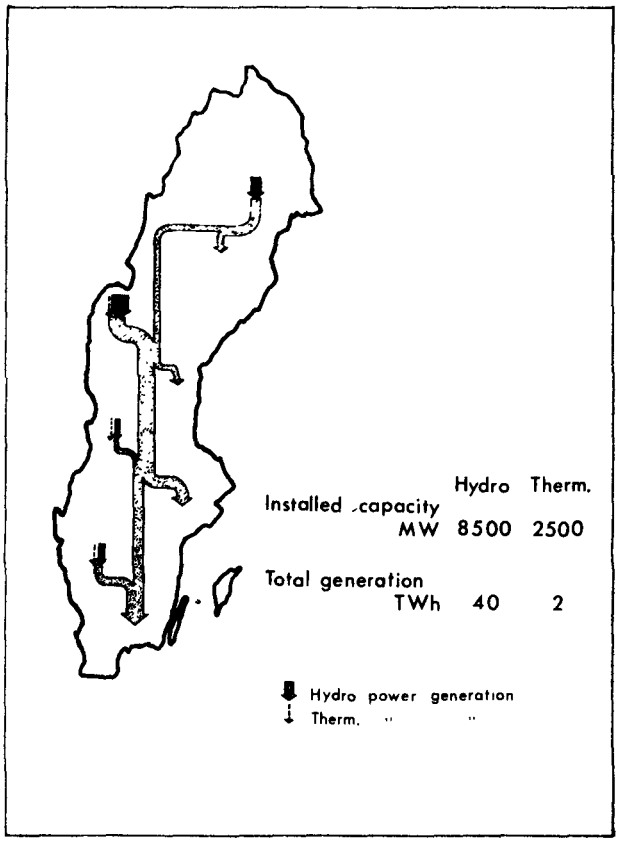


Figure 1. Power generation and distribution, Sweden 1963

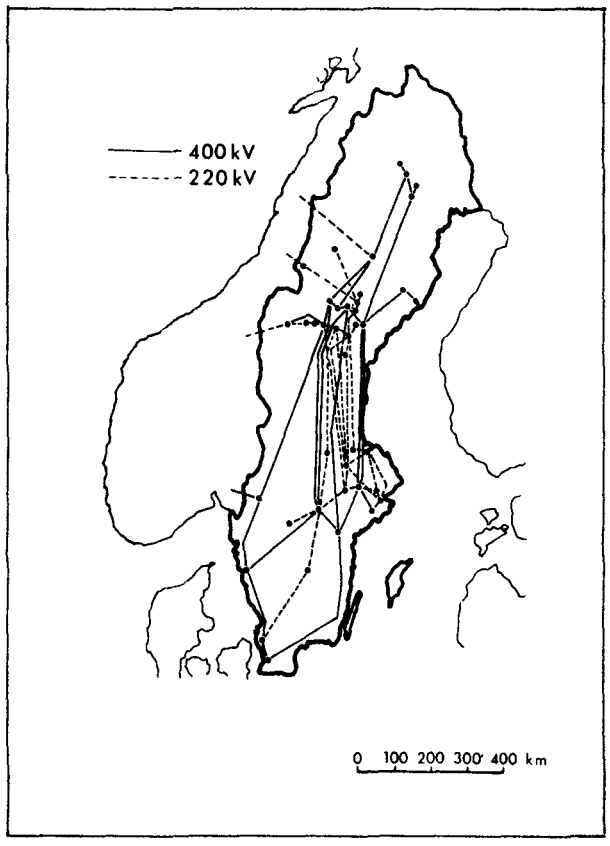


Figure 2. The Swedish high voltage transmission system, 1970

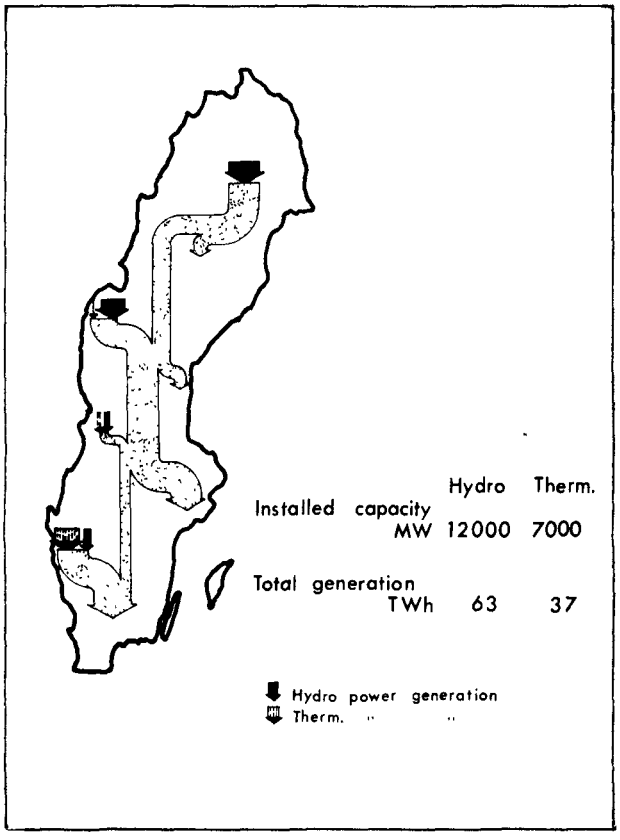


Figure 3. Power generation and distribution, Sweden 1980

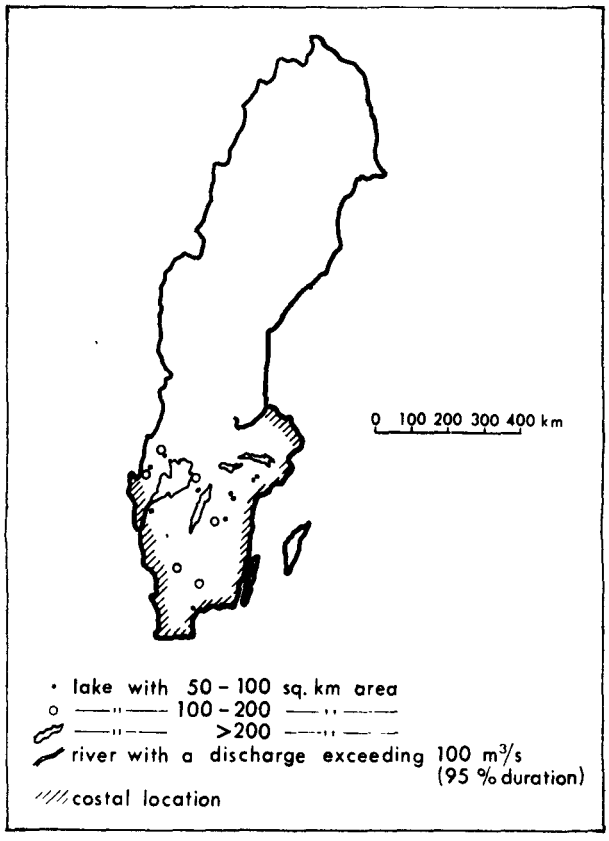


Figure 4. Availability of cooling water in southern Sweden

As to *station size*, a limit of 1 000 to 1 200 MW has been estimated for the period under consideration, when load distribution and transmission costs are taken into account.

Transmission costs

For rough estimates, the transmission costs on the high voltage system for a distance of 100 km can be calculated to be about \$1.00/kW+1.2% losses. The present value over the lifetime for a 1 000 MW station will then be of the order of 15–20 million dollars or \$15–20 per installed kW.

In most cases, however, a complete analysis of the whole transmission network has to be undertaken in order to determine the influence of station location on transmission costs.

OTHER FACTORS AFFECTING THE SITING OF GENERATING STATIONS

Environmental factors may affect the optimization of station size and location, or they may change the cost comparison between conventional and nuclear plants. The more important of such environmental factors are the following:

(i) Cooling water supply;
(ii) Fuel transportation and storage;
(iii) Hygiene and safety considerations;
(iv) Amenity considerations.

Cooling water supply

The necessary cooling water requirements for a 1 000 MW conventional steam power station are about 40 m^3/s if a temperature increase of 8 °C is allowed. For a nuclear plant of current design, some 70 m^3/s would be required, due to lower thermal efficiency. If the cooling water is to be taken from a river, the low water flow in the river has to be: 80 m^3/s for a conventional plant and 140 m^3/s for a nuclear plant. Half of the low water flow is assumed to be by-passed through the station, giving an average temperature increase of 2–3 °C downstream of the plant, taking the heat exchange from water surface to air into account. Alternatively, a lake or an artifical storage basin could be used. In that case a minimum of 50 km^2 area would be necessary.

As can be seen from Fig. 4 there are two rivers and about 20 lakes in southern Sweden that fall within the required category. The best siting possibilities are, however, presented by coastal locations, using seawater for cooling.

In the absence of an adequate cooling water supply, one would have to adopt an arrangement with cooling towers, resulting in a cost penalty. This latter can be estimated to be between \$15 and 20/kW installed capacity for a 1 000 MW station.

These figures refer to dry, closed-cycle cooling towers. For wet towers the cost increase is somewhat lower, but they would require evaporation of about 1 m^3/s water which may not be desirable.

Fuel transportation and storage

Sweden has practically no fossil fuel of its own. In 1963, the total import of oil was 16 million tons, of coal 3 million tons. Most of the fuel is used for heating purposes, at present only about 2% is consumed for power generation.

The freight rates have a decisive influence on the cost of fuel at the plant site. The present cost of heavy fuel oil delivered by large tankers c.i.f. Swedish west coast port corresponds to about \$1.30 per Gcal (\$0.30 per million Btu).

Recently made clearances of the approaches to the Baltic will make this price valid for large draught east coast ports also. For locations where transfer to lighter tonnage is necessary, a cost increase in the order of 10% would be incurred.

Plant sites other than coastal cannot be economicaly justified, due to the heavy cost penalty of land transportation.

According to governmental regulations, a minimum reserve of fuel must be kept in storage, corresponding to about one-third of the annual consumption. The actual storage capacity will therefore amount to about 50 % thereof, representing an investment of about \$5 per installed kW.

For the above-mentioned destinations coal cannot compete with oil, its c.i.f. cost being substantially higher. Also the storage and handling facilities constitute a rather heavy cost item. A 1 000 MW coal-fired plant, for instance, would require about 100 000 m^2 storage area, and the ash would need about the same. This is reflected in the plant capital cost, which is in the order of 20% higher than for an oil-fired plant. The present tendency in Sweden is to design plants exclusively for oil-firing.

Nuclear stations do not require large-tonnage harbour facilities. During operation, the spent fuel casks will constitute the heaviest items shipped, with weights in the order of 20–100 tons. These may be designed for both road, railway and ocean shipping.

Hygiene and safety

The amount of air pollutants from a 1 000 MW station using different fuels is illustrated in Table 1.

Health criteria for different pollutants are at present not comparable. Several countries have adopted permissible levels of contamination by sulphur oxides and other combustion products. These permissible

Table 1. Airborne waste per year from a 1 000 MW(e) station

	Conventional		Nuclear
	Oil	Coal	Water moderated and cooled
SO_2 (tonnes)	90 000	75 000	—
NO_x (tonnes)	14 000	14 000	—
Dust (tonnes)	1 500	1 000–5000 [a]	—
Radioactivity, representative values	1 mc Ra	3–30 mc Ra	^{131}I: millicuries ^{85}Kr: curies T: kilocuries
Free radicals, gas-ions	?	?	?

[a] Depending on filter efficiency.

values are, however, closely related to actual concentrations in large industrial districts and only an order of magnitude or so below what is considered as somatically harmful.

Pre-treatment of oil to remove the sulphur is today considered too costly, but has been recommended [8]. Flue gas washing is considered costly and gives loss of buoyancy of gases due to reduced temperature. Dry-absorption of pollutants shows some promise for the future.

In radiological work, permissible values have been laid down on purely theoretical grounds, using extrapolations from experimental data on acute high dose rate exposures. For environmental concentrations one usually bases the permissible values upon genetic, not somatic considerations. Agents present in relases from fossil fuel plants also undoubtedly have genetic effects. However, these have, been studied much less or not at all, and no quantitative estimates have been internationally agreed upon. This difference in calculation of permissible concentrations could very well approach a factor of 1 000 representing the degree of over-caution in radiological norms relative to those for combustion products.

Safeguards against the environmental effects of extraordinary occurrences are more or less unknown outside the nuclear field. Oil storage, gas works or plants storing large amounts of chlorine are, for instance, not banned from city limits, even though they represent potential hazards at least as large as those of nuclear reactors to life and health.

Nuclear power siting is complicated by rigid and conservative siting regulations. If these practices should be unnecessarily restrictive, which seems most probable, this will distort our evaluation in the choice between the different power sources. One may, however, assume that investments in safety devices and selection of inherently safe reactor designs together with an appropriate containment shall make it possible to locate nuclear reactors, where they ave most suitable for reasons other than those of safety.

Amenity considerations

A new law has recently been proposed in Sweden regulating the licensing system for all types of industry that create offensive or disturbing emissions. This will evidently make it easier to have the two types of generating sources compared on equal grounds.

Property damage due to corrosion alone by pollutants is very large, of the order of 200 M$/yr in Sweden, a large part of which ought to be charged to fossil fuel combustion. No such problems occur with nuclear power plants.

FEASIBLE SITES

Preliminary surveys based on the factors mentioned show that there are at least 10–15 sites along the coastline of southern Sweden that without excessive costs could be used for the location of a large thermal power station of at least 1 000 MW(e) capacity. If only nuclear power is considered, the number of acceptable locations might be larger.

As far as fossil fuelled plants are concerned, inland locations can be excluded for reasons given above. For nuclear plants, inland siting is feasible. Taking into consideration the composition of the actual power system, however, inland locations with sufficient cooling water supply would induce penalties in transmission costs for a large part of the additional thermal capacity needed during the period considered. Thus, at least for the next 3–4 plants to be built, coastal siting will be chosen whether they are conventional or nuclear.

It can be foreseen that in the future safety considerations will not exclude a nuclear station where a given location is favourable for a thermal plant. Therefore, a situation may present itself where the choice between a conventional and a nuclear plant can be markedly influenced by factors of hygiene determined by normal operation. In the following, the relevant health criteria will be discussed in more detail.

HEALTH CRITERIA

Non-radiological health criteria

Background

In Scandinavia, investigations of air pollution have been conducted in Oslo, Gothenburg, Stockholm [3], Helsinki and at a few other places. There have not been any acute smog episodes in Scandinavia like the ones in London, and photo-chemical smog as in Los Angeles has not been encountered.

One report [3] states, that the effect of continuous exposure to low level air pollution is much more important hygienically than short, but intense, smog episodes. Among the diseases of the respiratory organs that have shown a statistical correlation with air pollution, lung cancer has been most investigated. This disease has increased in frequency considerably, especially in England and the USA. Prindle [4] has shown that the mortality rates for cancer of the trachea, bronchus and lung increase markedly with city population size in the United States. The mortality rates for white males due to lung cancer has been shown to differ by 14 deaths per 100 000 inhabitants a year between large cities and rural areas [5]. This difference appears to be proportional to the measured air pollutants. There might not always be a straightforward relationship between pollution and disease. A primary reduction resistance may show up as an aggravation of some already existing disease or lead to ageing effects and life shortening.

Quantitative, realistic estimates of this death toll through air pollution can scarcely be made in Sweden due to the low level of pollution and the few data available. However, as in other countries there exists a correlation between incidence of male lung cancer and city population size.

The winter-time level of SO_2 in central Stockholm [3] is around 4–6 pphm (parts per hundred millions). This is about one-third of the London and about one-half of the New York average level. On one occasion (inversion, March 1963), it reached the average level in London (17–20 pphm). The average concentration of SO_2 over a 24 h period in the American National Air Sampling Network (50 stations) was 3 pphm, ranging from 140 to 1 pphm [6].

Soot concentrations are also low even during inversions, 0.07 mg/m³. A high value in London in December 1957 was 0.32 mg/m³. Experience and laboratory experiments have shown that there exists synergism between SO_2 and soot particles. It is evident that the air pollution in Sweden is not very pronounced and excludes smog episodes and photo-chemical smog. It is especially important to note that the 3, 4-benzopyrene content in Stockholm and Oslo is very low. The median value in Stockholm during winter is 0.001 μg/m³. In comparison, the median annual concentrations in English cities lie within the range 0.013 to 0.046 μg/m³ [7].

Environmental criteria

As indicated above, recommendations will probably soon be issued on permissible emissions in Sweden. The authors intend to discuss only the basis for a proposal of criteria for sulphur dioxide. Maximum allowable SO_2 ground level limits have now been established or recommended in different countries. (Table 2). These levels are supposed to be low enough to be without any phytotoxic effect or in offensive or deleterious to sensitive people. Very strict requirements are laid down in the Soviet Union, where only such concentrations as produce no indicatable changes

Table 2. Maximum permissible sulphur dioxide ground level limits

Country	MAC (pphm)	Total cumulative daily exposure time (h)
USA (San Francisco Bay Area Air Pollution Control District) [9]	20	No limit
	30	8 (day time)
	100	1 (" ")
	150 (max)	0.62 (" ")
England [10]:		
Heavily industrialized district or high housing density	34	24
All others	28	24
West Germany [11]	20	24
	30	0.5
Soviet Union [12]	8	24
	23	0.33

(reactions of the central nervous system) are permissible. Thus, the values vary according to the basis of standardization. It is striking to note how close the permissible levels are to what is actually experienced in the large cities of the respective countries. It would be wise to make use of the favourable situation in Sweden and establish MAC-levels for continuous exposure that are rather close to the winter mean value of central Stockholm. It is therefore suggested that a value in the order of 5 pphm should be chosen. The general background level lies around 0.5–1 pphm in smaller communities, and it does not seem unrealistic to limit the additional contribution from any stack plume to about 4 pphm at ground level.

Recent investigations have shown that the hazards of SO_2 are not merely a product of time and concentration [12]. For shorter integrated periods than 24 h, the following equation for permissible concentrations

has been used, where the progressive effects from concentrations above the long-time value are compensated for [9]:

$$C_1 = C_0 + \frac{K}{t}$$ where t = exposure time, h

C_1 = permissible concentration, ppm

$C_0 = 0.05$ ppm

and $K = 0.8$ ppm h.

An investigation [13] sponsored by AB Atomenergi and the Swedish State Power Board has analysed the diffusion climate in Sweden. According to this, different sampling times Δt give a variation in the average concentration $K(\Delta t)$ over the sampling time according to

$$K(\Delta t) = K(1\text{ h})\,(\Delta t)^{-0.40} \quad (\text{for } 0.01 < \Delta t < 1\,000\text{ h}).$$

This equation indicates that one should compare short-time permissible levels with values applicable for periods even longer than 24 h. Figure 5 shows the relationship between time and safe concentrations of

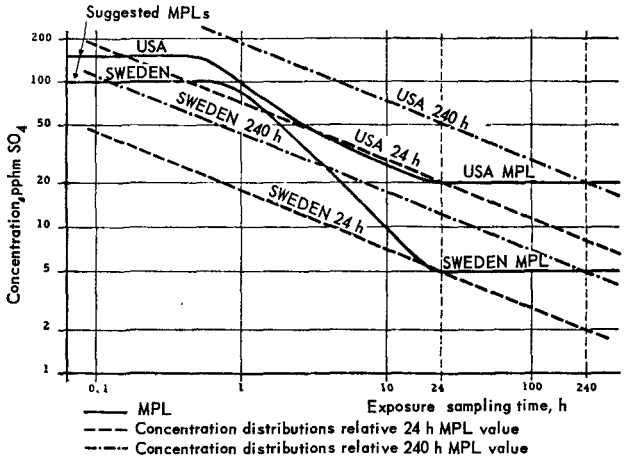


Figure 5. Suggested maximum permissible levels for Sweden and USA compared with normal concentration distributions from Swedish investigations [13]

SO_2 as suggested in the USA and Sweden. The short-time concentration should be limited to 100 pphm in Sweden.

For both relationships it is shown how atmospheric diffusion may vary according to the Swedish investigation. One curve is drawn using a 240 h interval as a standard for a long-time average, the orther using a 24 h interval.

American investigations have shown that the average pollution level during a sampling time of 5 min may be at the most 5 times higher than the average over 24 h, which is also in accordance with the values presented by Nonhebel [10]. The Swedish values show a factor of around 10 between 24 and 0.1 h integration intervals, and a factor of 20 between 240 and 0.1 h intervals.

Radiological health criteria

Background

Both the United Nations and other international agencies [14–16] have been instrumental in creating universally accepted rules for permissible exposures to ionizing radiation. There is very little real difference in the regulations applied all over the world, the fundamental assumptions, medical and experimental evidence being principally the same. It was mentioned earlier that there is a basic difference between radiological and all other health criteria. Only the former try to consider genetic effects, and even assume these to be one of the limiting factors for permissible releases to the environment. At present one must accept this situation as establishing an extra safety margin for nuclear plants or, possibly, an unknown, extra hazard from the operation of fossil fuel plants.

Environmental criteria

The ICRP [15] recommends a dose to people living in the neighbourhood of nuclear plants below 500 mrem per year, about twice the background dose rate in Sweden.

Exposure from nuclear power plants may reach people through external radiation from plant or transports, through liquid waste releases or through ventilation air and off-gas releases to the air. External exposures are well controlled through shielding requirements. No significant part of permissible doses need to be apportioned for this type of exposure.

Liquid waste releases may vary with the type of reactor, how it is operated and if large maintenance operations have to be performed on site. Limits for such releases are calculated taking possible biological transport mechanisms into consideration. Experience from reactors in other countries and calculations for Swedish conditions show that, if a sufficient cooling water supply can be provided, this may also serve as a recipient for the low level active waste. High and medium level waste may be stored or processed on site, and will not constitute any hazard for the environment. It may be possible to release all active waste from plants located on the west or Baltic coasts and only make provisory arrangements for storage of emergency quantities. Inland sites will probably always need equipment for storage of high activity waste, but releases of low level activity from normal operation and maintenance will not give excessive exposures even if made to relatively small recipients.

Releases of activity to the atmosphere are very small for most present-day power reactors. They could vary

by orders of magnitude depending upon the reactor system in question and the air and off-gas cleaning equipment installed. A radio-nuclide of interest is iodine-131. Release of ^{131}I is limited by the concentration in the air-grass-cow-milk-child chain. Releases will vary depending upon how much fission product activity is released from the fuel. Source strengths in the vicinity of 0.01 μc/s may have to be accepted. This corresponds to an air flow of over 10 000 m³/s, and is easily obtained with a moderate stack height. While of low radio-toxicity, tritium emanating from D_2O reactors could be of interest, as it is not economically practical to separate tritiated water from common water vapour in the release. The maximum possible rate of tritium release from a D_2O moderated and cooled reactor will be of the order of 0.001 c/s, corresponding to 2 000 m³/s of air contaminated to what is permissible off-site.

In comparison, SO_2 releases from large fossil fuel plants correspond to over 10 000 000 m³/s of air contaminated to 5 pphm. This will require large smoke stacks and/or large distances from popultaion centres.

While short-time exposures may be critical for SO_2 exposure, even when long-time average levels are acceptable (Fig. 5), this is not so for the releases to the atmosphere from nuclear plants. Permissible average conditions there correspond to 0.05 mrem/h. The ICRP gives no limitations for short-time exposures, but Swedish authorities recommend that radiological workers do not exceed 1 mrem during any second. The ICRP recommends only not to exceed 3 000 mrem per quarter year. The former recommendation would permit 360 mrem, the latter 3 000 mrem over 0.1 h. The available margin over the permitted long-time average is here a factor of 100 000, while for SO_2 it was less than 100. Thus, even very unusual weather situations could never give critical increases in the short-time dose rates from activity releases, as long as the long time averages are acceptable.

Suggested criteria

Referring to the above discussion, we find that the requirements for releases from nuclear plants under normal operations are negligible. Thus, criteria for these releases are not economically critical, even though the ICRP and equivalent recommendations are conservative.

SUMMARY

In this paper the more important power economy and environmental factors in connection with the siting of power plants have been discussed. Qualitative analyses and rough quantitative estimates of the economic influence of the various factors have been made. As a result, it has been found that for sites that fulfil the other requirements, the choice between nuclear and fossil fuelled plants can be greatly influenced by health criteria during normal operation. A review of existing criteria, both radiological and non-radiological, has been made, and it is shown how the former present safety margins against somatic effects that are several orders of magnitude higher than the latter. Nevertheless, the radiological health criteria will not exclude any location that fulfills other siting criteria. As regards non-radiological health standards, such are to a great extent non-existent in Sweden. Suggestions for criteria regarding emission of SO_2 to the atmosphere have been presented.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/611 Suède

Facteurs d'économie, de sûreté et d'hygiène régissant l'implantation de différents types de grandes centrales thermiques en Suède

par S. O. W. Bergström et C.-E. Holmquist

Les économies de dépenses en capital que permettent de grandes installations sont plus élevées dans le cas de centrales nucléaires que dans celui de centrales classiques. Aussi la tendance vers des centrales thermiques très grandes aura-t-elle bientôt pour effet que le choix entre les deux sources d'énergie sera conditionné surtout par des facteurs d'hygiène et de sûreté.

Donc, avant même que l'énergie nucléaire soit concurrentielle, il est indispensable pour un pays de procéder à une analyse et à une évaluation comparative des critères de santé publique et de sûreté relatifs aux sources d'énergie possibles. L'emploi de combustibles fossiles a entraîné des pertes regrettables de vies humaines et causé des dégâts et de l'incommodité dans l'environnement. Dans le cas de l'industrie nucléaire, par contre, les critères en matière de choix du site et de sûreté sont souvent exagérés et arbitraires.

En Suède, les centrales thermiques seront, pour des raisons dues à la situation des centres de charge, implantées dans les régions méridionales du pays au cours des 10 à 15 années à venir. Du point de vue stratégique, des centrales nucléaires souterraines situées dans l'intérieur du pays sont désirables. La rareté de sites à l'intérieur du pays, avec des sources d'eau de réfrigération suffisamment grandes pour des centrales d'une puissance de l'ordre de 1 000 MW(e) ou plus, et les frais d'investissement additionnels entraînés par la construction de centrales souterraines de cette importance minimisent fortement les possibilités d'implantation au centre du pays. Les grandes centrales classiques doivent être situées sur la côte en raison des facilités de transport du combustible, puisqu'en Suède la totalité des combustibles fossiles est importée.

Des études préliminaires indiquent qu'il existe en tout et pour tout 10 à 15 emplacements le long de la côte méridionale de la Suède qui pourraient offrir les installations portuaires, les moyens de transport et l'eau de réfrigération nécessaires pour de grandes centrales thermiques. Une centrale de 2 000 MW(e) utilisant du charbon exige une superficie de stockage de l'ordre de 150 000 m^2 pour le charbon et environ la même superficie pour les cendres et scories. Environ un tiers de la consommation annuelle doit être stocké conformément aux règlements officiels en vigueur. On ne doit pas non plus oublier les dégagements de poussière, le bruit, la pollution des eaux et les dangers d'incendie résultant de la manutention de ces combustibles. Les problèmes de transport et de stockage d'une centrale nucléaire semblent, d'autre part, être relativement faibles.

Le choix de l'emplacement de réacteurs nucléaires en Suède a, en principe, fait ressortir l'importance du facteur distance. Pour les centrales nucléaires futures d'une puissance supérieure à 500 MW(e) le choix d'un emplacement peut se heurter à des difficultés, même dans les régions côtières faiblement peuplées. Il est de la plus haute importance que cet obstacle soit surmonté par l'emploi de réacteurs à sûreté inhérente et l'application de mesures de sécurité limitant les risques.

Les centrales thermiques utilisant du charbon ou du mazout n'ont pratiquement aucune possibilité de réduire la pollution de l'air par les particules résultant de la combustion et les oxydes de soufre et d'azote. Par comparaison, la dose à laquelle est soumis le public du fait de la radioactivité émise par un réacteur nucléaire est négligeable et toujours prévisible d'avance; elle reste toujours sous le contrôle du préposé à la surveillance, même en cas d'accident majeur.

En dépit des déversements inévitables d'une grande centrale nucléaire vers les eaux environnantes, le facteur principal affectant ces masses d'eau sera leur réchauffage après passage dans le circuit de réfrigération, et c'est là un problème commun à tous les types des centrales.

A/611 Швеция

Вопросы безопасности, гигиены и экономические факторы при выборе участка для расположения различных типов крупных тепловых электростанций в Швеции

С. О. В. Бергстрём *et al.*

При строительстве крупных электростанций снижение капитальных затрат будет бо́льшим для атомных, чем для обычных электростанций. Кроме того, при строительстве очень крупных тепловых электростанций выбор между этими двумя источниками энергии определяется главным образом проблемами гигиены и безопасности.

Таким образом, еще до достижения атомными электростанциями конкурентоспособности основной проблемой для страны становится ана-

лиз и определение на равных условиях критериев охраны здоровья и безопасности населения, позволяющих сделать выбор между двумя источниками энергии. Использование ископаемых видов топлива сопровождается прискорбной гибелью человеческих жизней, не говоря уже о неудобствах, причиняемых населению прилегающего района. С другой стороны, при использовании атомных электростанций критерии безопасности, определяющие выбор места строительства, иногда являются произвольными и слишком завышенными.

В течение ближайших 10—15 лет тепловые электростанции в Швеции ввиду местонахождения районов нагрузки будут строиться на юге страны. Однако по стратегическим соображениям целесообразно строить подземные атомные электростанции во внутренней части страны. Недостаточное количество крупных потребителей электроэнергии для станций электрической мощностью порядка 1000 *Мвт* во внутренней части страны и дополнительные капитальные затраты, связанные со строительством подземной электростанции, значительно уменьшают возможность строительства таких станций во внутренней части страны. Крупные электростанции обычного типа необходимо строить у побережья для сокращения расходов на транспортировку топлива, поскольку все виды ископаемого топлива Швеция импортирует из заграницы.

Предварительные обследования показывают, что на южном побережье Швеции имеется около 10—15 участков, расположенных поблизости от портов и имеющих транспортную сеть и охлаждающую воду, необходимые для крупных тепловых электростанций. Тепловая электростанция электрической мощностью 2000 *Мвт*, работающая на угле, должна иметь складскую площадь для хранения запаса угля, равную 150 000 $м^2$, и примерно такую же площадь для складирования золы. В соответствии с существующими положениями такая электростанция должна иметь запасы топлива в размере одной трети своего годового потребления. Не следует забывать также о пыли, шуме, загрязнении воды и опасности пожаров, характерных для тепловых электростанций. С другой стороны, для атомной электростанции проблема транспортировки и хранения топлива является сравнительно небольшой.

В основе практической деятельности по выбору участков для ядерной электростанции в Швеции до сих пор основное значение играл фактор удаленности. В будущем при строительстве крупных атомных электростанций с агрегатами электрической мощностью более 500 *Мвт* трудности могут возникнуть даже в малонаселенных прибрежных районах Швеции. Для решения этой проблемы первостепенное значение имеет использование ядерных реакторов, имеющих безопасную конструкцию, что приведет к ограничению принимаемых мер предосторожности.

У тепловых электростанций, работающих на нефтяном и угольном топливе, отсутствуют практически эффективные средства для уменьшения загрязнения воздуха в результате сжигания серы, азота и окислов этих элементов. Для сравнения можно указать, что доза облучения населения в результате работы атомной электростанции является пренебрежимо малой; ее всегда можно заранее определить, и она находится под контролем оператора станции даже в случае крупной аварии.

Несмотря на неизбежный сброс отходов из атомной электростанции в водный проток, основное влияние на этот водный проток будет оказывать повышение температуры охлаждающей воды, а этот фактор является одинаковым для электростанций всех типов.

A/611 Suecia

Los factores de seguridad, higiene y economía en la ubicación, en Suecia, de grandes centrales térmicas de distintos tipos

por S. O. W. Bergström y C. E. Holmquist

Al aumentar la capacidad, los ahorros en los costos de inversión de las centrales nucleares son mayores que los correspondientes a las clásicas. Por otra parte, la tendencia actual hacia centrales térmicas muy grandes hace que la selección entre los dos tipos de fuentes de energía dependa mucho de los factores de higiene y de seguridad.

Por consiguiente, cuando la energía nuclear sea competitiva con la de otras fuentes, o incluso antes de que eso suceda, es un requisito fundamental el que los países analicen y determinen, en términos de igualdad, los criterios de higiene y de seguridad públicas de las distintas fuentes de energía. Por el empleo de combustibles fósiles ha habido que pagar, desafortunadamente, vidas humanas además de los daños e inconvenientes producidos en los alrededores de la central. Por otra parte, los criterios de seguridad y de emplazamiento empleados en la industria nuclear son a veces muy ambiciosos y arbitrarios.

A causa de la distribución de la demanda, las centrales térmicas a instalar en Suecia, dentro de los 10 ó 15 años próximos, se ubicarán en el Sur. Sin embargo, por razones estratégicas, sería conveniente la ubicación subterránea en el interior del país. La escasez de emplazamientos interiores idóneos para centrales de 1 000 MW(e), unido a las inversiones adicionales que lleva consigo la construcción de las centrales subterráneas, hace que la ubicación en el interior sea poco probable. Las grandes centrales de combustible fósil deben ubicarse en la costa a causa de las facilidades de transporte, ya que en Suecia se importa todo el combustible fósil utilizado.

Análisis preliminares indican que, en total, existen de 10 a 15 emplazamientos a lo largo de la costa meridional de Suecia con las instalaciones portuarias, de transporte y agua de refrigeración que se necesitan para instalar centrales térmicas grandes. Una central de carbón de 2 000 MW(e) necesita un parque de carbón de unos 150 000 m^2 de superficie y una extensión casi igual para las escorias. De acuerdo con las autoridades competentes, se debe tener almacenado, en cualquier momento, un tercio del consumo anual. No deben despreciarse ni el polvo, el ruido, la contaminación del agua ni los riesgos de incendio de todas estas operaciones. Por otra parte, comparativamente los problemas del transporte y almacenamiento en el caso de una central nuclear parece que son pequeños.

Las normas seguidas en Suecia para ubicar instalaciones nucleares han insistido, en principio, en la importancia del factor distancia. En las grandes centrales nucleares, con unidades de más de 500 MW(e), a instalar en el futuro, pueden presentarse dificultades de emplazamiento aun considerando las regiones costeras menos pobladas. Es muy importante que se pueda salvar este obstáculo utilizando reactores intrínsecamente seguros y con salvaguardias que limiten los riesgos.

En las centrales que queman petróleo o carbón no se pueden emplear métodos prácticos que reduzcan la contaminación del aire con las partículas de la combustión y con los óxidos del azufre y del nitrógeno. En comparación, la dosis de radiación a que está expuesto el público como consecuencia de la explotación de una central nuclear es despreciable, se puede predecir siempre y está en manos del operador de la central, aun en el caso de un accidente importante.

A pesar de que no se puede evitar el que el agua de refrigeración de las centrales nucleares descargue en las masas de agua disponibles, el factor más importante que influye en el recipiente es el aumento de la temperatura del agua de refrigeración, lo que es común a todos los tipos de centrales.

Release of fission gases in reactor coolants under normal operation

By K. G. Vohra *

The large scale use of reactors for power production would require the assessment of long term effects of small release of fission gases in the environments during normal operations. The presence of fission gases in reactor coolants under normal operation has been reported in literature [1, 2]. The mechanism of release from the perfectly clad fuel into the coolant has not been fully understood. The fissions from the traces of uranium that may exist on the outside of fuel cladding would give a small release of fission gases as well as solid fission products in the coolants. The larger rates of discharge of fission gases over the solid fission products may possibly be caused by permeation from the rich source of fission gases in the core through the cladding by some form of diffusion. The latter has been studied by the measurement of fission gases ^{88}Kr and ^{138}Xe in the cooling water in a swimming pool reactor. The change in relative activities of the two fission gases during the time of drift across the cladding has been studied to distinguish between the release by fission recoil due to the surface contamination of the cladding by uranium and the release from the core through the cladding.

DIFFUSIONAL RELEASE VERSUS SURFACE RECOILS

The fission gases krypton and xenon constitute a very important part of the fission products produced inside the reactor fuels. Approximately one fission gas atom is produced for every four fissions. The fission gas activity as well as pressure inside the fuels build-up with the time of operation. The activity of most of the fission gases reaches an equilibrium value after a few days of operation except for ^{85}Kr (half life 10.3 years) and ^{133}Xe (half life 5.27 days). At equilibrium time the total activity of fission gases in the fuel is 5.6×10^5 curies per megawatt of power. The pressure build-up would correspond to the release of approximately 3 cm^3 of fission gases at NTP per megawatt day. This is mainly due to stable fission gases which have a much higher yield in fission than the radioactive gases. The build-up of pressure would depend on the void space available inside the fuels and between the fuel and claddings for different types of fuel elements. The pressure build-up does not attain equilibrium, as is the case with radioactivity build-up. The permeability of fission gases through the cladding materials by some form of diffusion at the normal operating temperatures and the pressures attained inside the fuels may therefore be likely.

On the other hand, the fissionable atoms present as impurity within the recoil range of fission fragments in the cladding materials do give rise to direct release of fission gases, the degree of release depending on the concentration of the impurity. The rate of release is given by the product of fission yield and the rate of fission within the recoil range. If diffusion is the predominant mode of release, the rate of release would sharply depend on the radioactive half life because the shorter-lived fission gases would show larger decay during the time of drift through the cladding.

The diffusion of gases through metals can take place by the following processes; namely, substitutional diffusion, interstitial diffusion and grain boundary or void diffusion. The first two processes represent true diffusion in the classical sense whereas the last process depends on structural properties of the metal such as grain size, porosity and microscopic defects in the metal. Within the fuels the fission gas atoms are distributed homogeneously and may be considered to be in an abnormal state of solution. The fission gas atoms are known to diffuse through the uranium lattice as the temperature is increased [3-5]. This phenomenon may be considered to provide a constant source of fission gas atoms at the fuel-cladding interface. This source has the fission gases in equilibrium except for ^{85}Kr, the number of atoms of each gas isotope being proportional to its fission yield.

The likely mechanism of permeability of inert fission gases through the claddings is surface adsorption and dissolution followed by diffusion and desorption from the other surface. Larger rates of penetration through the cladding may also be caused by changes in the structural properties of the claddings caused by long term irradiation and natural porosity in some types of cladding materials.

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THEORETICAL CONSIDERATIONS OF RELEASE FROM A CLADDED FUEL PLATE

We shall consider the diffusion formulae applicable to the case of MTR fuel plates. This is mathematically the simplest case in which diffusion in one direction only can be considered, because the thickness of the fuel plates is very small compared to the surface area. The solutions of the Fick's diffusion equations have been applied to the release of fission gases in a swimming pool reactor using this type of fuel plate, to study the possibility of diffusion taking place through the cladding. The MTR fuel plates consist of a sandwich of enriched uranium fuel between two thin plates of aluminium cladding, with slight curvature. There is a sound metallurgical bond between the fuel core and the cladding. During irradiation in a reactor the fission gases are generated within the fuel. Large increase in the concentration of fission gases at the fuel-cladding interface is not likely, the actual concentration depending on the diffusion coefficients of fission gases in uranium and aluminium. No diffusion through the cladding is likely if the coefficient of diffusion of inert gases is smaller for aluminium than for uranium. The values of diffusivity of aluminium for inert and noble gases reported in literature are not smaller than those for uranium, at the same temperature, and may perhaps be larger.

We may therefore consider a constant concentration of fission gas atoms at the fuel-cladding interface, the atoms lost by permeation through the cladding being compensated by the atoms formed in fission. Fick's second law of diffusion for the accumulation of atoms from the interface into the cladding may be applicable.

According to this law, the concentration gradient in the cladding is given by

$$\delta C/\delta t = D\,(\delta^2 C/\delta x^2),$$

where C is the concentration of atoms at time t at a distance x in the cladding. For a constant source the boundary conditions are

$$C\,(0, t) = C_0 = \text{constant for all values of } t,$$
$$C\,(x, 0) = 0 \qquad \text{for } t = 0.$$

With the above boundary conditions the following solution of the equation can be obtained:

$$C\,(x, t) = C_0\,[1 - \operatorname{erf}\,(x/^2\sqrt{Dt})]. \tag{1}$$

The above solution has been widely used for similar problems. At time $t = \infty$, $C\,(x, t) = C_0 = \text{constant}$.

The rate of entry of atoms into the cladding per unit area is found from Fick's first law. If m is the number of atoms entering a unit area in a time t, according to Fick's first law

$$\delta\, m/\delta\, t = -\,D\,(\delta\, C/\delta\, x).$$

If Q is the number of atoms passing an area S of the cladding plate in time t the material flow is given by

$$\mathrm{d}\,Q/\mathrm{d}\,t = -\,S\,D\,(\delta\, C/\delta\, x).$$

Differentiating Eq. (1) with respect to $x/2$ we get

$$\mathrm{d}\,Q/\mathrm{d}\,t = -\,S\,C_0\ \sqrt{D/\pi t}\,\mathrm{e} - x^2/4Dt. \tag{2}$$

The time required by the fission gas atoms to diffuse across a thickness h of the cladding can be found from the following simple relation based on Eq. (2), with the approximation that the exponential term in the equation gives a value of 0.99, which implies that the time elapsed is large enough for equilibrium to be established between the rate of entry of the atoms into the cladding and the rate of exit, i.e., a steady state flow has been attained:

$$t = h^2/0.04\,D. \tag{3}$$

The approximation used in the above relation is valid for very small values of h and large t. This means that, after enough time has elapsed, the material flow at $x = h$ is nearly the same as at $x = 0$, with the boundary condition $C\,(h, t) = 0$ for all values of t. Value of t can be found from the change in the ratio of number of atoms of two different isotopes (with different half lives) during travel through the cladding.

ACTIVITY BUILD-UP IN THE COOLANT

The build-up of activity in the reactor coolant will take place in the case of release by fission recoils from the cladding surface as well as in the likely case of release by diffusion. The degree of build-up will depend upon the rate of release and the type of cooling system used. The two extreme cases are the boiling water reactor and the swimming pool reactor. In the former the fission gases are released within a very short time of their entry into the coolant water and there is practically no build-up. In the latter the build-up of fission gases and their daughter products takes place till equilibrium is established. We shall consider the build-up in a swimming pool reactor in more detail. In the case of recoils the build-up is given by

$$N = (f_1 y/\lambda + L_1)\left(1 - \mathrm{e}^{-(\lambda + L_1)T}\right), \tag{4}$$

where N is the number of atoms accumulated in the coolant in time T, and f, y, λ and L_1 are fissioning rate at the cladding surface, yield of the fission gas isotope, its decay constant, and fractional loss of atoms through coolant replacement, respectively. In the case of diffusional release the build-up is given by

$$N = (f_2 y/\lambda + L_1)\left(1 - \mathrm{e}^{-(\lambda + L_1)T}\right)\mathrm{e}^{-\lambda t}, \tag{5}$$

where t depends on the thickness of cladding and the coefficient of diffusion, as given by Eq. (3). f_2 is the

fissioning rate at the interface i.e., the fissions leading to the flow of atoms through the cladding.

Table 1 gives the number of atoms of important fission gases present in the coolant at equilibrium for an assumed rate of 10^4 fissions per second on the surface of the cladding, assuming 50% recoils entering the coolant. Table 2 gives the numbers of atoms in the coolant at equilibrium for 10^6 fissions per second at the interface, for $t = 10^2$, 10^3, and 10^4 seconds. A comparison between the theoretical ratios of any two selected fission gas isotopes from Tables 1 and 2 and the ratio observed experimentally would show whether the release is predominantly by recoils or by some diffusion-like mechanism. The apparent diffusion coefficient can be found from the value of t close to the experimental ratio, knowing the cladding thickness h.

Table 2. Activity of important fission gas isotopes in the coolant based on the diffusion model for different values of t for 10^6 fissions per second at the fuel-cladding interface

Fission gas isotope	λ (s^{-1})	Yield (%)	Activity $N\lambda$ $t = 10^2$/s	$t = 10^3$/s	$t = 10^4$/s
^{87}Kr	1.48×10^{-4}	2.77	2.74×10^4	2.38×10^4	6.3×10^3
^{88}Kr	6.9×10^{-5}	3.77	3.73×10^4	3.51×10^4	1.89×10^3
^{89}Kr	3.7×10^{-3}	3.98	2.74×10^4	1.0×10^3	—
^{90}Kr	2.1×10^{-2}	4.07	4.95×10^3	—	—
^{91}Kr	7.1×10^{-2}	3.17	2.56×10^1	—	—
^{133}Xe	1.5×10^{-6}	6.15	6.15×10^4	6.13×10^4	6.1×10^4
^{135}Xe	2.1×10^{-5}	6.00	6.0×10^4	5.89×10^4	4.88×10^4
^{137}Xe	3.0×10^{-3}	5.65	4.2×10^4	2.8×10^3	—
^{138}Xe	6.8×10^{-4}	5.74	5.35×10^4	3.37×10^4	6.65×10^1
^{139}Xe	1.7×10^{-2}	4.79	8.76×10^3	—	—
^{140}Xe	3.8×10^{-2}	3.33	7.5×10^2	—	—

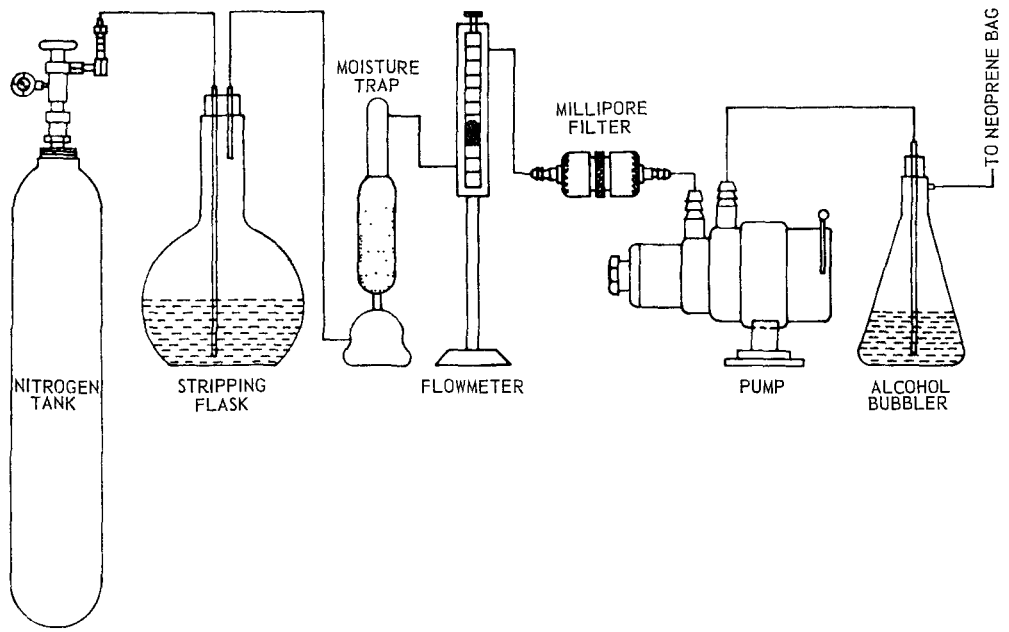


Figure 1. Apparatus used for stripping water samples for krypton-88 and xenon-138 measurement

Table 1. Number of atoms produced by fission and activity of important fission gas isotopes in water from surface recoils at equilibrium for 10^4 fissions per second on the cladding surface

Fission gas isotope	λ (s^{-1})	Yield (%)	Number of atoms (N)	$N\lambda/^2$
^{87}Kr	1.48×10^{-4}	2.77	1.87×10^6	138.5
^{88}Kr	6.9×10^{-5}	3.771	5.46×10^6	188.5
^{89}Kr	3.7×10^{-3}	3.98	1.08×10^5	199
^{90}Kr	2.1×10^{-2}	4.07	1.94×10^4	203.5
^{91}Kr	7.1×10^{-2}	3.17	4.46×10^3	158.5
^{133}Xe	1.5×10^{-6}	6.15	4.1×10^8	307.5
^{135}Xe	2.1×10^{-5}	6.00	2.86×10^7	300
^{137}Xe	3.0×10^{-3}	5.65	1.88×10^5	282.5
^{138}Xe	6.8×10^{-4}	5.74	8.45×10^5	287
^{139}Xe	1.7×10^{-2}	4.79	2.82×10^4	239
^{140}Xe	3.8×10^{-2}	3.33	8.76×10^3	166.5

FISSION GASES IN APSARA WATER

APSARA is a swimming pool type of reactor. This reactor has been operating continuously at a power of 300 kW, except for very short periods of shut-down. Total activity of fission gases in the core at equilibrium during continuous operation is estimated to be 1.68×10^5 curies. The pool contains 2.05×10^5 litres of water. 600–800 litres of water are added daily to compensate for the loss of water.

For the study of the mechanism of production it was necessary to select two gases with different half lives. It was also necessary to select gases with solid descendents of convenient half life as the method used depends on the daughter product activity measurement. ^{88}Kr and ^{138}Xe satisfied the above requirements.

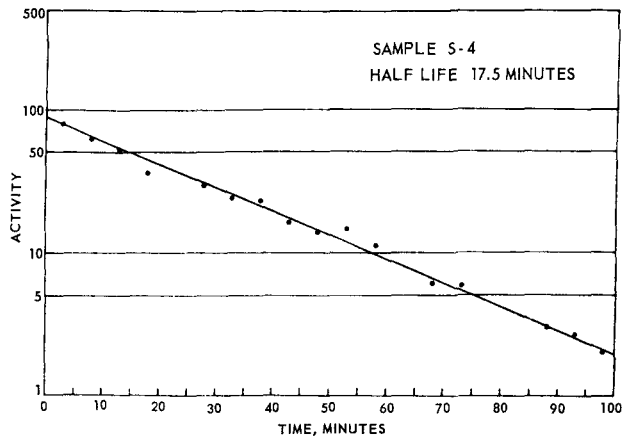


Figure 2. A typical decay curve of rubidium-88 from krypton-88 recovered by stripping of APSARA water

Figure 1 shows the apparatus used for stripping the water samples by a nitrogen stream and pumping the stripped gas into a neoprene bag through an alcohol bubbler. A detailed description of this type of stripper is given elsewhere [6]. The following procedure is followed for the estimation of ^{88}Kr and ^{138}Xe.

For ^{88}Kr a 500 cm^3 sample of water is collected in the stripping flask after the reactor has been in continuous operation for a few days. The sample is stored in the flask for 100 minutes to allow ^{138}Xe (half life 17 min) to decay to a negligible fraction. Stripping is carried out for 30 minutes with a nitrogen stream at 7 litres per minute. The carrier gas containing ^{88}Kr is filled in the neoprene bag through the alcohol bubbler and stored for 50 minutes to allow the growth of ^{88}Rb (half life 17.5 min) to nearly maximum value. The gas is filtered through a millipore filter paper. ^{88}Rb activity collected on the filter paper is measured with a beta counter. ^{88}Kr concentration in the water is estimated from ^{88}Rb activity. Figure 2 gives a typical decay curve for ^{88}Rb from stripped ^{88}Kr.

For ^{138}Xe the stripping of the sample is started 5 minutes after removal from the reactor. Stripping is carried out for 20 minutes with a nitrogen stream at 7 litres per minute. The storage time in the neoprene bag is 20 minutes. ^{138}Xe concentration is estimated from the beta activity of ^{138}Cs (half life 33 minutes).

A fraction of the fission gases released in the reactor pool water escapes into the air. The ratio R of the atoms released in the water to atoms released in the air is found from the following relation:

$$R = [A_w (\lambda + L_1)]/[A_a (\lambda + L_2)], \qquad (6)$$

where A_w and A_a are the total number of atoms of a gas in the water and air, respectively, found experimentally; L_1 and L_2 are the fractional rates of change of water and air, respectively.

For the measurement of airborne ^{88}Kr and ^{138}Xe, 5 400 litres of air are passed through a millipore filter paper, using a suction rate of 90 litres per minute and a suction time of 60 minutes. The daughter product activity of ^{88}Rb and ^{138}Cs collected on the millipore filter paper is measured at two different times T_1 and T_2. The concentrations of ^{88}Rb and ^{138}Cs in the air are estimated by solving two simultaneous equations for activity on the filter paper at T_1 (15 minutes) and T_2 (30 minutes). From the total activity of the daughter products in the reactor hall at equilibrium the rates of injection of ^{88}Kr and ^{138}Xe are computed, taking into account the loss of activity through ventilation. The denominator in Eq. (6) above is actually this rate of injection into the air. These rates are found to be 6.94×10^7 atoms/minute for ^{88}Kr and 3.1×10^7 atoms/minute for ^{138}Xe. These rates are also calculated for water from ^{88}Kr and ^{138}Xe activities measured. The ratio R is found to be nearly 116 for ^{88}Kr and 11.5 for ^{138}Xe. The total emission rates are calculated from the concentrations of ^{88}Kr and ^{138}Xe in water and the ratios R.

Table 3 gives the activities of ^{88}Kr and ^{138}Xe in the water at equilibrium, for water samples collected from different parts of the pool. The ratio of ^{88}Kr/^{138}Xe has been given in the last column of the table. ^{88}Kr activity in the pool ranges from 6 400 to 40 000 pc/litre. The average value is 17 800 pc/litre. The corresponding average value for ^{138}Xe is 790 pc/litre. The contribution of recoils from the surface arising from uranium contamination has been estimated to be of the order of 500 pc/litre for ^{138}Xe and 320 pc/litre for ^{88}Kr. This estimate is based on the measurement of ^{95}Zr in water, an isotope having a very short lived gaseous precursor and not likely to come from the core. ^{88}Kr release is larger by a factor of 20 to 125 than the estimated recoil

Table 3. Activities of krypton-88 and xenon-138 in the pool water at different points in the pool, and the ratios krypton-88/xenon-138

Sampling point [a]	^{88}Kr (pc)	^{138}Xe (pc)	^{88}Kr/^{138}Xe ratio
S1	2.79×10^4	4.4×10^2	63.2
S2	3.03×10^4	7.0×10^2	43.2
S3	1.50×10^4	10×10^2	15
S4	2.56×10^4	4.6×10^2	55.5
S5	4.00×10^4	9.5×10^2	42
M1. . . .	2.20×10^4	7.4×10^2	29.7
M2. . . .	0.78×10^4	10×10^2	7.8
M3. . . .	1.19×10^4	11.4×10^2	10.4
M4. . . .	0.77×10^4	10×10^2	7.7
M5. . . .	0.64×10^4	8.2×10^2	7.8
B1	1.00×10^4	4.3×10^2	23.2
B2	1.36×10^4	7.6×10^2	17.9
B3	1.39×10^4	13.5×10^2	10.3
C1	1.74×10^4	3.0×10^2	58
Mean. . .	1.78×10^4	7.9×10^2	22.6

[a] S: sampling points near the surface of the pool.
M: sampling points near the middle of the pool.
B: sampling points near the bottom of the pool.
C: sampling point close to the core.

release, whereas in the case of ^{138}Xe the release is larger only by a factor of 2 or so and in a few cases ^{138}Xe release is of the same order as expected from recoils.

The total release of ^{88}Kr and ^{138}Xe has been estimated from the total activity in the pool water and the total activity in the air corrected for ventilation. ^{88}Kr activities are estimated to be 3.65×10^9 pc in water and 3.15×10^7 pc in the air. The corresponding ^{138}Xe activities are 1.62×10^8 pc in water and 1.41×10^7 pc in the air. The ratio of total ^{88}Kr released to total ^{138}Xe released is nearly 21. In the core the ^{138}Xe activity is larger than ^{88}Kr and the above ratio is 0.65. Large change in the ratio of $^{88}Kr/^{138}Xe$ activity may be attributed to the origin of these activities in the reactor core. It is seen from Table 2 that $^{88}Kr/^{138}Xe$ activity ratio is 28.2 for $t = 10^4$ seconds. The above experimental value of the ratio will also correspond to a value of t of the same order.

If the experimental value of t, i.e., 10^4 seconds, is substituted in Eq. (3) for the known value of $h = 5 \times 10^{-2}$ cm (cladding thickness) a surprisingly high value of the diffusion coefficient is obtained. It is therefore concluded that the normal process of diffusion is not operative and some other form of permeability through the cladding is taking place. The release mechanism may, however, be explained on the basis of the diffusion model for the sake of simplicity.

DISCUSSION

The presence of large concentrations of the fission gas isotope ^{88}Kr in the APSARA pool water cannot be explained on the basis of release by fission recoils resulting from uranium contamination on the surface of the aluminium cladding. The release by the above mechanism will give, at equilibrium, concentrations of different fission gases in the pool proportional to their fission yields. The fission yield of ^{138}Xe is nearly 1.6 times that of ^{88}Kr. The measured average concentration of ^{138}Xe in the water is smaller than that of ^{88}Kr by a factor of 22.6. The variations in this factor as seen in Table 3 may be attributed to the lack of proper mixing in the pool.

The possibility of pinholes in the cladding is ruled out because this would give rise to direct leakage of fission gases and the ratio of ^{88}Kr to ^{138}Xe will not decrease sharply as has been observed. The same argument would apply to the leakage from the joints in the cladding frame. The other likely causes are: permeability of fission gases through microscopic cracks in the cladding surface, and through pores in the normal structure of aluminium.

The operating temperatures of the fuel in APSARA is normally below 100 °C. At this temperature the coefficient of diffusion through the aluminium crystal may be expected to be extremely small. Tobin [7] has reported a diffusion coefficient of 10^{-8} cm^2/s for xenon diffusion in silver at 800 °C after the gas was introduced into the metal surface by bombarding the metal with ionised gas. Using a similar technique Murray and Pincus [8] obtained a value of 10^{-9} cm^2/s for the diffusion coefficient of helium in aluminium at 470 °C. The latter study was carried out to investigate the possibility of coolant contamination problem in the likely case of diffusion of fission gases through cladding materials. Very large values of diffusion coefficient in the present investigation suggest that at low operating temperatures permeation may take place either through normal porosity of the cladding materials or through microscopic defects. In the process of manufacture of MTR fuel plates the claddings are tested for microscopic cracks by X rays. However. surface cracks can result by prolonged irradiation in reactors. In a recent investigation at Brookhaven National Laboratory, Castleman and others [9] found that under irradiation at 500 °C microscopic cracks were formed on the aluminium surface giving rise to higher release of iodine in a diffusion experiment.

The present study suggests the likelihood of small release of fission gases from the fuel core into the coolant in the reactors using aluminium as the cladding material. If the release is considered in terms of a diffusion mechanism the time of drift across the cladding is found to be of the order of 10^4 seconds for cladding thickness of 5×10^{-2} cm (0.020 inches). For a given cladding thickness the degree of release of any fission gas isotope will be sharply dependent on its half life. The comparatively longer lived fission gases such as ^{85}Kr, ^{133}Xe, $^{133}Xe^m$ and ^{135}Xe are likely to penetrate even thicker claddings if the mechanism of permeability is not peculiar to very thin claddings. Similar investigations in different types of reactors would provide valuable data for the study of the phenomenon.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/790 Inde

Dégagement de gaz de fission dans le réfrigérant d'un réacteur en fonctionnement normal

par K. G. Vohra

Pendant l'exploitation normale des réacteurs utilisant différents types d'éléments combustibles, des gaz de fission se dégagent dans le réfrigérant qui est en contact direct avec les éléments combustibles par le mécanisme de diffusion gaz-métal. Des expériences entreprises pour mesurer la fraction de gaz de fission libérés dans le réfrigérant dans un réacteur piscine ont fourni des renseignements précieux sur ce mécanisme.

Les mesures de gaz de fission et de leurs produits de filiation dans le réfrigérant et dans l'air atmosphérique dans un réacteur piscine ont établi le mécanisme de diffusion proposé selon lequel l'ampleur du dégagement de gaz de fission dépendrait de la nature et de la composition du combustible, de sa température et de son aire superficielle, de la géométrie de l'élément combustible et du coefficient de diffusion. Le modèle proposé peut servir à calculer le dégagement de gaz de fission dans des réacteurs de types différents dans les conditions normales d'exploitation.

A/790 Индия

Выделение газообразных продуктов деления в теплоносителе при нормальной эксплуатации реактора

К. Г. Вохра

В процессе обычной эксплуатации реакторов с различными типами тепловыделяющих элементов газообразные продукты деления выделяются в теплоноситель, находящийся в непосредственном контакте с тепловыделяющими элементами в результате диффузии газа в металле. Эксперименты, проведенные по оценке доли газообразных продуктов деления, попавших в теплоноситель в погружном реакторе, позволили получить ценные данные об этом механизме.

Измерения газообразных продуктов деления и их дочерних продуктов в теплоносителе и в атмосферном воздухе для погружного реактора позволили определить предполагаемый механизм диффузии, в котором степень выделения газообразных продуктов деления зависит от характера и состава топлива, его температуры, площади поверхности, геометрии тепловыделяющих элементов и коэффициента диффузии. Предложенная модель может быть использована для расчетов выделения газообразных продуктов деления в реакторах различных типов при нормальной эксплуатации.

A/790 India

Escape de gases de fisión del refrigerante durante el funcionamiento normal del reactor

por K. G. Vohra

Durante el funcionamiento normal de los reactores que utilizan diferentes tipos de elementos combustibles, se desprenden gases de fisión en la parte del refrigerante que se halla en contacto directo con los elementos combustibles, por un mecanismo de difusión gas-metal. Los experimentos realizados por los autores con el propósito de calcular la fracción de gases de fisión que pasan el refrigerante en un reactor tipo piscina han proporcionado una valiosa información sobre el citado mecanismo.

Las mediciones de los gases de fisión y de sus productos descendientes, efectuadas tanto en el refrigerante como en el aire ambiente, en un reactor tipo piscina, han permitido proponer un mecanismo de difusión con arreglo al cual la cantidad de gases de fisión desprendidos dependería de la naturaleza y composición del combustible, de su temperatura y superficie, de la geometría del elemento combustible y del coeficiente de difusión. El modelo que se propone puede utilizarse para el cálculo de la cantidad de gases de fisión desprendidos en reactores de distintos tipos, en régimen normal de funcionamiento.

Primary cooling water system characteristics under transient conditions in the Canada-India Reactor

By V. Surya Rao, N. Veeraraghavan, M. S. R. Sarma, G. G. Joshi and M. Sampathkumaran *

The Canada-India Reactor [1] at Trombay is a 40 MW natural uranium fuelled, heavy water moderated, light water cooled, research reactor employing sea-water as the secondary coolant. The uranium is in the form of long cylindrical rods. Each rod is sheathed in an aluminium tube having three fins for centering in an outer sheath, the annulus thus formed serving as passage for the light water coolant. While the design of the CIR has followed closely that of the NRX reactor at Chalk River, the major difference between the two reactors lies in the cooling water system. The latter has a once through system while the former has a closed loop recirculation system. This has been made necessary since the large quantities of fresh water required for cooling are not available at the CIR site.

Under normal operating conditions, light water coolant is recirculated through fuel rods in a closed loop at a flow rate of 4 200 gal(imp)/min by four 300 hp pumps (Fig. 1). The light water enters and leaves the fuel rods at pressures of 176 and 34 psig respectively. On loss of power to the pumps, the reactor is designed to scram on any of the following parameters of the primary cooling water system: *(a)* low flow through individual fuel rods, *(b)* low differential pressure across the reactor and *(c)* low gross flow through the reactor. Simultaneously the normal recirculating system is automatically converted to a once-through gravity feed emergency system with a gross flow rate of 1 200 gal(imp)/min, achieved in about 8 s. The emergency flow path is from the emergency water storage tank, through two parallel check valves through the reactor and to the dump tanks via two parallel, air operated, quick opening valves. Suitable check valves and an inverted U loop are provided in the circuit to prevent draining the remainder of the primary system when the emergency system is operating.

During the period beginning with the loss of power to pumps and until the emergency flow has been fully established, the flow coast down in the recirculating loop must provide sufficient cooling to prevent fuel failure. This transition period is most important in a hazards review since it is during this period the reactor power and coolant flow will be decaying at different rates. This paper presents the studies that were carried out to establish the flow during this transition period and the associated temperatures of the coolant and the aluminium cladding.

HYDRODYNAMIC CHARACTERISTICS OF THE PRIMARY COOLING SYSTEM

Prior to commissioning and testing the system, a theoretical study was carried out to predict flow coast down due to the inertia of the system on pump trip. Assuming that all the kinetic energy stored in the system is available to do work, the following equation [2] was derived, representing the flow as a function of time:

$$F = \frac{F_0}{\frac{D_0}{2E_0}t + 1}$$

where F is the system flow at time t,
F_0 is the system flow at zero time,
D_0 is the energy dissipation rate at zero time,
and E_0 is the energy stored in the system at zero time.

Zero time refers to the instant of pump trip.

The shape of the curve is shown in Fig. 2.

FIELD TESTS DURING COMMISSIONING

During the commissioning period, flow coast down tests were carried out using fast acting recording equipment to measure the following parameters simultaneously: cooling water gross flow (main and emergency), pump discharge pressure, pump suction pressure, and pump rpm, at the instant or time of loss of power to pumps. The location of the instruments is marked in Fig. 1. Cooling water pressures in the upper and lower headers were extrapolated from the pump discharge and suction pressures.

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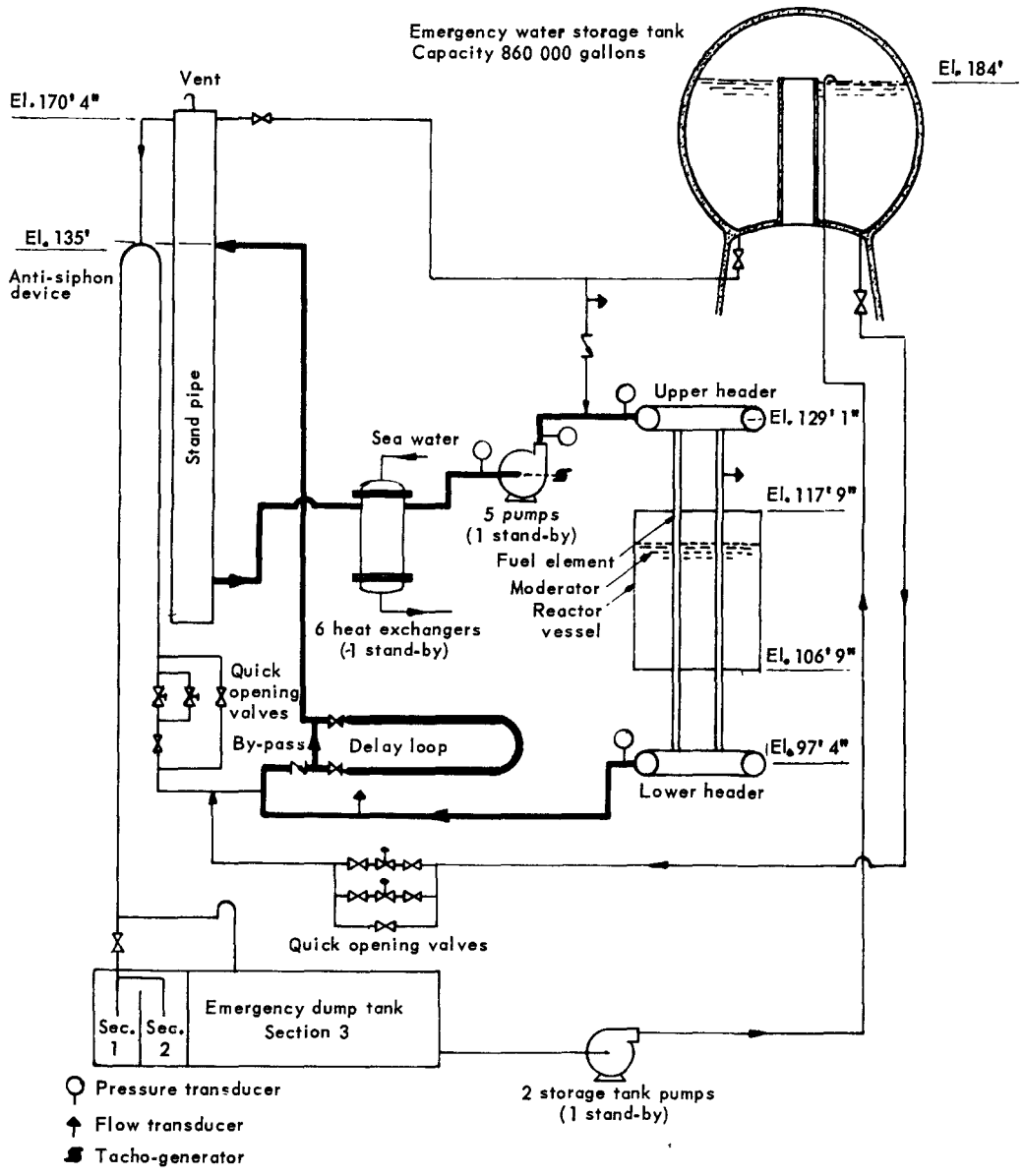


Figure 1. Simplified flow diagram of the primary cooling water system

From the test results, the following observations were made: *(a)* flow continues at the equilibrium rate for approximately 200 ms following pump trip; *(b)* flow during coast down was pulsating; and *(c)* pressures measured at the above locations in the system also revealed pulsations.

No satisfactory explanation could be given for the flow remaining constant for the first 200 ms. However, by a shift of the time axis by 200 ms and with a suitable adjustment of the constant D_0/E_0 in the above equation, the theoretical flow coast down was in general agreement with the field results. This was the basis for all further evaluations. In view of the observed flow and pressure oscillations, it was feared that at certain stages of the coast down the fuel might be without adequate cooling, which would result in nucleate boiling in the coolant channels and solidus or bulk boiling at the fuel exit, leading to fuel failure.

Thus at this stage it was found necessary to establish that the theoretical flow coast down curve could meet the requirements of adequate cooling and therefore a computer programme was detailed to study this. The programme required computation of aluminium cladding and saturation temperatures along the length of the fuel rod, and bulk coolant temperatures for different types of fuel rods (with varying cooling water annuli), at different power outputs and different operating heavy water heights (different specific power) using the theoretical flow coast down curve and the theoretical power decay curve (Fig. 2).

There is an inherent time delay in the system between the loss of power to pumps and the start of

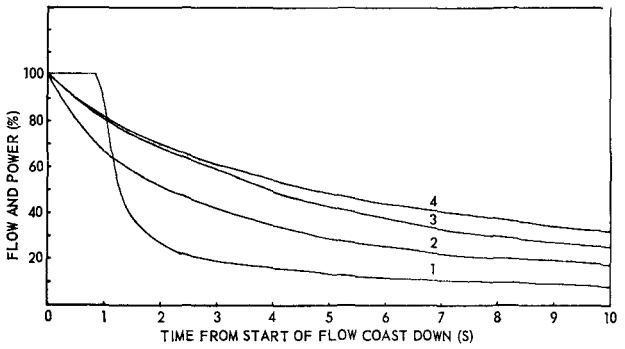


Figure 2. Transient characteristics of flow coast down and power decay

1: Power decay; 2: Flow without flywheels; 3: Flow demand curve; 4: Flow with flywheels

reactor power decay. This is the cumulative effect of two factors: *(a)* the time delay between initiation of pump trip and initiation of reactor scram due to the characteristics of the parameters of the coolant system initiating reactor scram and *(b)* the time taken for the scram devices to become effective in lowering reactor power, once the scram has been initiated. Out of the parameters that initiate reactor scram, the gross flow is the last and is taken into consideration here to give the most pessimistic estimate. The above cumulative time is estimated at 1.08 s. Since the flow remains constant for the first 200 ms, the net flow-power time delay is taken as 880 ms in all the computer programmes.

Results of a typical computer run for the aluminium cladding and coolant temperatures during the coast down are shown in Fig. 3 for a 275 kW rod under aged conditions i.e., when the coolant flow is reduced. Figure 4 shows the saturation temperature of the coolant at the hottest spot of the fuel rod taking into consideration the pressure transients. As can be seen from the curves (Figs. 3 and 4), even though the saturation temperatures obtained on the basis of the theoretical flow coast down curve were within acceptable limits, the temperatures corresponding to the observed pressure oscillations indicated nucleate boiling (and or bulk boiling for higher rod power outputs) during the transient period. It can be stated that on a pump trip, as the flow coasts down and as the pressure decays in the water annulus of the fuel rod, the safest condition for operating the reactor would be that in which the cladding temperature of the fuel at any instant is lower than the saturation temperature corresponding to the water pressure in the annulus at that particular instant.

Reactor operation at 40 MW involves rod power outputs in the region of 330 kW. Pending further investigations, the maximum rod power was limited to 300 kW and the maximum reactor power to 30 MW (as the fuel charge was fresh and thus had more flow than in the aged condition).

INVESTIGATIONS LEADING TO 40 MW OPERATION

With a view to assessing the safety of reactor operation at high powers and incorporating any modifications required for the primary cooling system, a series of tests, as detailed below, were conducted.

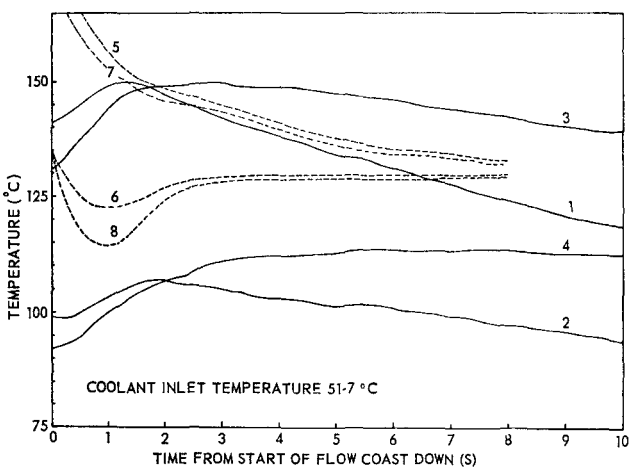


Figure 3. Cladding and coolant transient temperatures

1: Cladding temperature at hottest spot, 330 kW aged rod, with flywheels; 2: Coolant outlet temperature, 330 kW aged rod, with flywheels; 3: Cladding temperature at hottest spot, 275 kW aged rod, without flywheels; 4: Coolant outlet temperature, 275 kW aged rod, without flywheels; 5: Saturation temperature at hottest spot, with flywheels; 6: Saturation temperature at lower header, with flywheels; 7: Saturation temperature at hottest spot, without flywheels; 8: Saturation temperature at lower header, without flywheeis

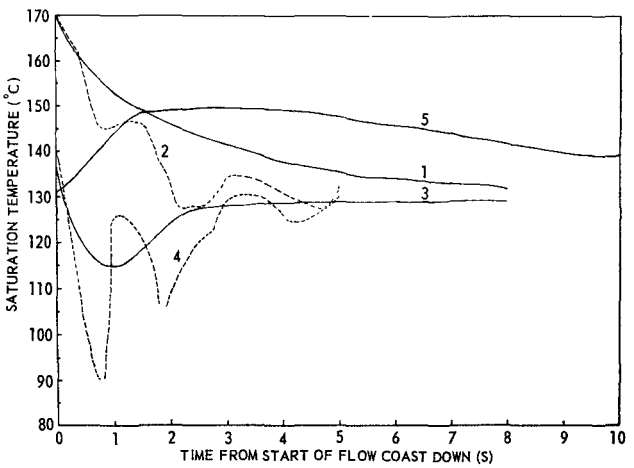


Figure 4. Saturation temperatures associated with pressure transients observed during commissioning tests

1: Saturation temperature at hottest spot without pressure oscillations; 2: Saturation temperature at hottest spot with pressure oscillations; 3: Saturation temperature at lower header without pressure oscillations; 4: Saturation temperature at lower header with pressure oscillations; 5: Cladding temperature at hottest spot for a 275 kW aged rod

Flow coast down tests

Since flow coast down tests during the commissioning stages revealed large pulsations in the flow and pressure decay curves, it was considered necessary to repeat earlier tests conducted to check the reproduceability of the pulsations, before any further investigations or modifications were carried out. In addition to the parameters measured previously, upper and lower header pressures were also measured. When these tests were repeated, no pulsations were observed in the flow and pressure decay curves. No reasonable explanation could be offered for the absence of pulsations. It was also noticed that the flow coast down was not instantaneous with respect to the time of pump trip, a time delay of around 200 ms being noticed as in earlier tests (Fig. 2). The observed flow coast down was in good agreement with the theoretical prediction. Even so, since rod outputs in the region of 330 kW would be encountered for reactor operation at 40 MW, it was feared that the flow coast down would not be adequate to suppress nucleate and or bulk boiling on the basis of the indication from the 275 kW rod. Therefore a computer programme was arranged to obtain a flow demand curve meeting the above requirements. This curve is also shown in Fig. 2. From this, it was clear that certain modifications had to be made to the system to improve the flow coast down characteristics to better or equal this demand curve before the reactor power could be raised to 40 MW.

By-passing the delay loop

To prevent deposition of corrosion products on the fuel rods, it was found necessary, at this stage of the investigation, to by-pass the delay loop (Fig. 1) thereby eliminating about 60% of the mild steel surface exposed to the cooling water. Calculations for the flow coast down showed that the kinetic energy contribution of the delay loop to the total energy of the system was approximately 4 %, and as such, system flow coast down characteristics would not be affected significantly by eliminating the delay loop. These were confirmed by flow coast down tests.

In addition, since certain modifications were to be made to improve the flow coast down of the system, it was thought that the kinetic energy loss due to by-passing the delay loop could be incorporated in the modifications without compromising any safety features. On the above considerations, the delay loop was by-passed.

Modifications to the system to improve flow coast down

The following alternatives were considered for improving the system flow behaviour on a pump trip:

(a) A high header tank of about 550 ft head and of adequate capacity riding on the system to provide normal cooling by gravity feed after pump trip;

(b) A pressurised tank also riding on the system to meet the same requirements as *(a)*;

(c) The installation of flywheels on the pump motor shafts.

Installation of a high header tank was not taken up in view of the expenditure and the time involved in completing the project, involving extended reactor shut-down for connecting into the system.

Neither was the installation of a pressurised tank considered desirable in view of the complexity of controls necessary to make this system operative and again the long reactor shut-down involved. Although these two modifications would provide normal flow rates even after pump trip, this was not considered necessary since the requirements of the flow demand curve could be met by the installation of flywheels. This scheme could be effected with least effort and expenditure and what was more important, without causing any extended reactor shut-down.

INSTALLATION OF FLYWHEELS ON MOTOR SHAFTS

To meet the requirements of the flow demand curve, flywheels (30 in diam and 2 in thick mild steel plate) were installed, one on each pump motor shaft. After each installation, flow coast down tests were repeated to ensure that the flow decay pattern was improving as expected, before the reactor was operated. Figure 2 shows the flow decay pattern with all flywheels installed. The improvement in flow rate during coast down is quite marked, being 30%. Installation of flywheels also resulted in the improvement of coolant pressures with corresponding increase in saturation temperatures during coast down.

ANALYSIS OF THE TEST RESULTS

Flow-power time delay

The time delay between the flow decay and power decay is important in computing the transient temperatures on a pump trip; the larger the time delay, the greater the magnitude of the transient temperatures. Hence it was considered essential to obtain these values by experiment with the reactor operating at low power. The results showed that this delay was always in the region of 500 ms.

Subsequent to the above findings, the control circuit was modified to initiate reactor scram directly on loss of power to the pumps. This improvement reduced the flow-power delay time to about 200 ms, which is the minimum achievable with the existing design of scram devices. Nevertheless the value of

880 ms was used in all further computer work, since (i) it is a pessimistic value and (ii) it facilitates comparison with the earlier computer results of the commissioning tests.

Computer studies after the installation of flywheels

Using theoretical flow coast down data with flywheels, a theoretical power decay curve and a flow power delay time of 880 ms, computer studies were carried out to assess the safety of fuel rods on pump trip at 30 MW operation. A typical computer result giving cladding and bulk coolant temperatures for a 330 kW rod under aged conditions (hottest fuel rod at 40 MW with near full loading condition) is shown in Fig. 2. Also shown are the transient temperatures which would be obtained for 1 275 kW aged rod prior to installation of flywheels. Whereas the 275 kW rod case was in the region of nucleate boiling, the 330 kW rod case after flywheel installation was nowhere near this region. In fact, the saturation temperatures for this case were always higher than the cladding temperatures. Thus the marked improvement due to flywheels is clearly demonstrated.

The above computer studies also showed that a fuel rod with an output of 375 kW is quite safe from nucleate and bulk boiling considerations. This provides enough flexibility under different operating conditions since with a natural uranium loading of the reactor, an output of 375 kW for a fuel rod is not expected to be exceeded.

FUEL ROD OUTLET WATER TEMPERATURE MEASUREMENTS

Computer calculations on temperature transients showed that the outlet water temperature was gradually rising, reaching a maximum in about 3 or 4 s. It was decided to measure this temperature for the hottest rod, using a fast acting thermocouple (80 ms time constant) directly in the water stream. The reactor was operated at 4, 10, 15 and 20 MW each time before tripping the pumps. The resulting records showed no rise in bulk outlet water temperature up to 10 MW and a temperature rise of 0.5 to 2 deg C, when the reactor was operating at 15 and 20 MW. These tests were conducted before the flywheels were installed. Similar tests conducted after the installation of the flywheels and after modifying the pump control circuit for varying reactor powers up to 40 MW (fresh fuel rod with power outputs up to 330 kW) showed a negligible rise in temperature. The computer results indicated a rise of about 8 deg C in bulk outlet water temperature for an aged 330 kW rod (Fig. 2) after pump trip.

The discrepancy between the computer results and the observer values is believed to be due to the pessimistic assumption with regard to flow condition and flow-power delay time.

CONCLUSION

Installation of flywheels on primary cooling pumps considerably improves flow-rates through fuel rods during the change-over period to emergency cooling. This provides the necessary basis for operating the reactor at 40 MW without uncertainties regarding safety of fuel rods for different operating conditions.

ACKNOWLEDGEMENT

The authors take pleasure in acknowledging the valuable advice received from the Atomic Energy of Canada Limited at various stages of these studies, particularly in connection with the computer programme.

They are also grateful to Shri S. Srinivasan and other colleagues of the Reactor Operations Division for conducting the flow coast down tests on the fast acting recording equipment, and to Shri P. K. Mohan Rao for his assistance in arranging the computer programme.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/793 Inde

Caractéristiques du circuit primaire de refroidissement par eau du réacteur Canada-Inde dans des conditions transitoires

par V. Surya Rao *et al.*

Le circuit primaire d'eau de refroidissement du réacteur Canada-Inde est conçu pour la recirculation avec un débit normal de 4 200 gallons(imp)/min. Tout arrêt des pompes primaires a pour résultat un arrêt d'urgence du réacteur, la recirculation étant alors remplacée par un système à circuit ouvert avec alimentation de secours par gravité avec un débit de 1 200 gallons(imp)/min, atteint en 8 secondes. Cette période de transition entre la recirculation normale et le plein débit de secours est extrêmement importante dans

les évaluations des risques, puisque pendant cette période la diminution progressive *(coast down)* du débit et de la puissance devrait être telle qu'elle assure un refroidissement suffisant pour éviter la rupture du combustible.

Pendant la période de mise en service initiale du réacteur, nombre d'essais ont été entrepris pour déterminer le comportement du débit et de la pression du système lors du passage de la recirculation normale au régime en circuit ouvert. Les données expérimentales sur le *coast down* du débit s'accordent généralement avec les caractéristiques hydrodynamiques prévues du système, bien qu'on ait observé des oscillations des débits et des pressions. Un programme de calcul électronique a été établi pour calculer les températures moyennes du gainage d'aluminium et de la masse du réfrigérant (dans la période de transition) sur la longueur de la barre de combustible à l'aide des données théoriques sur le *coast down* du débit et la chute de la puissance. La machine à calculer était aussi programmée pour estimer les températures de saturation du réfrigérant et pour déceler toute ébullition dans ces conditions. En superposant les résultats des calculs aux oscillations de débit et de pression qu'on a observées dans les essais de mise en service, on est arrivé à une imprécision si grande que des conditions dangereuses d'ébullition nuclée ou en masse paraissaient possibles. En attendant des études complémentaires, la puissance du réacteur a été limitée à 30 MW et les puissances de barres à 300 kW.

On a envisagé plusieurs modifications au système de refroidissement pour obtenir de meilleurs débits et pressions pendant la période de transition. Des essais de *coast down* du débit ont été répétés pour estimer la reproductibilité des résultats précédents. En même temps, des mesures expérimentales sur les températures transitoires du réfrigérant à la sortie de la barre de combustible ont été faites dans des conditions de puissance moyenne du réacteur pour estimer la précision des prévisions de calculs. Aucune oscillation de pression ou de débit n'a été observée pendant ces essais de *coast down* du débit ni pendant les essais suivants.

Les températures observées étant toujours beaucoup plus basses que les valeurs prévues, il n'y avait de toute évidence pas de risque que le revêtement d'aluminium de la barre de combustible fonde ou que la température de transition de phase de l'uranium soit atteinte. Néanmoins, les températures observées indiquaient une ébullition nuclée pour la marche à 40 MW. On a donc conclu que le régime du réacteur à 40 MW exigeait certaines modifications au circuit de refroidissement primaire. L'amélioration des caractéristiques hydrodynamiques du système a été obtenue en installant des volants sur chacune des pompes du réfrigérant primaire, et on a répété les essais de *coast down* du débit. Un programme de calcul avec ces données indique l'absence de toute ébullition dans les barres de réacteur pour un niveau de puissance du réacteur correspondant à une puissance individuelle des barres de combustible ne dépassant pas 330 kW. Seule une légère ébullition nuclée fut observée à des puissances de barre de combustible de 375 kW. Des mesures expérimentales ultérieures des transitoires de température du gainage et de l'eau d'échappement n'ont, pendant la période de transition qui suit l'arrêt des pompes, pas ou guère indiqué d'augmentation par rapport aux valeurs normales d'équilibre pour les barres de combustible en marche à haute puissance. Comme les valeurs expérimentales et l'analyse par calcul électronique ont établi nettement qu'il n'y avait pas de danger à faire marcher le réacteur à puissance élevée, on l'a poussé au plein régime prévu, soit 40 MW, et l'exploitation a été satisfaisante.

A/793 Индия

Характеристики первичной водяной системы охлаждения в переходных условиях в Канадско-Индийском реакторе

B. Сурна Рао *et al.*

Первичная система охлаждения воды в Канадско-Индийском реакторе сконструирована для рециркуляции с нормальной скоростью потока при 19 068 *л/мин*. Любое выключение первичных насосов приводит к аварийной остановке реактора, причем рециркуляция превращается в одноходовую аварийную систему подачи самотеком со скоростью потока 5450 *л/мин* за 8 *сек*. Переходный период между обычной рециркуляцией и полностью аварийными условиями потока является самым важным при оценке опасности, поскольку именно в течение этого периода должен быть гарантирован спад потока и мощности, чтобы обеспечить достаточное охлаждение и предотвратить разрушение тепловыделяющих элементов.

Во время пуско-наладочных работ было проведено несколько опытов по определению поведения потока и давления в системе в переходном периоде от нормальной рециркуляции к одноходовой системе. Экспериментальные данные по спаду потока вообще согласуются с предсказанными гидродинамическими характеристиками системы, хотя наблюдались колебания потока и давления. Была разработана программа для вычисления на счетно-вычислительной машине температуры алюминиевой оболочки и общей температуры теплоносителя (за весь период изменения) вдоль топливного стержня с использованием теоретических данных по спаду потока и мощности. Для вычислитель-

ной машины составлялись также программы по оценке температуры насыщения теплоносителя и обнаружения любого кипения при этих условиях. Если колебания потока и давления, обнаруживаемые во время пуско-наладочных работ, накладывались на результаты, получаемые на счетной машине, ошибка была столь велика, что могли возникнуть небезопасные условия пузырчатого и (или) общего кипения. В связи с результатами дальнейших исследований мощность реактора была ограничена до 30 *Мвт* и мощность, отдаваемая каждым стержнем, до 300 *квт*.

Рассмотрены различные модификации в системе теплоносителя для улучшения скорости потока и давления в течение периода изменений. Испытания спада потока были повторены, чтобы можно было оценить воспроизводимость ранее полученных результатов. В то же время были проведены экспериментальные измерения температурных переходов теплоносителя на конце топливного стержня при работе реактора на промежуточной мощности. Эти измерения позволяют оценить точность предсказаний, сделанных при помощи счетно-решающей машины. Во время этих, а также последующих испытаний спада потока колебаний давления и потока не наблюдалось. Наблюдаемые температуры по существу много меньше предсказанных значений; было ясно, что нет риска расплавления алюминиевой оболочки топлива или достижения температуры фазового перехода для урана. Тем не менее наблюдаемые температуры свидетельствуют о том, что при работе на мощности 40 *Мвт* возможно возникновение пузырчатого кипения. На основании этого был сделан вывод, что для работы реактора на мощности 40 *Мвт* необходимо выполнить определенные модификации в первичном контуре теплоносителя. Усовершенствование гидродинамических характеристик системы было достигнуто за счет установки маховиков на каждый насос для теплоносителя первичного контура, после чего испытания по спаду потока были повторены. Результаты, полученные для этого случая на счетно-вычислительной машине, указывают на отсутствие кипения какого бы то ни было вида в топливных стержнях для условий работы реактора на мощности, когда мощность, отдаваемая отдельным топливным стержнем, не превышает 330 *квт*. При мощности отдельного стержня 375 *квт* наблюдалось небольшое пузырчатое кипение. Дальнейшие экспериментальные измерения температурных переходов для оболочки и воды на выходе показали небольшое превышение или не показали никакого превышения обычных равновесных значений для топливных стержней в условиях работы на большой мощности в течение периода изменения после выключения насоса.

Поскольку экспериментальные значения и анализ вычислений определенно подтверждают, что нет опасности для эксплуатации реактора на большой мощности, мощность реактора была увеличена до полного проектного значения 40 *Мвт*, причем реактор работал удовлетворительно.

A/793 India

Características del sistema primario de refrigeración por agua en condiciones transitorias, en el reactor Canadá-India

por V. Surya Rao *et al.*

El sistema primario de refrigeración por agua del reactor Canadá-India está diseñado para que se recircule un flujo normal de 20 m^3 por minuto (4 200 gal(imp)/min). Cualquier parada de las bombas primarias produce una parada rápida del reactor, en cuyo caso la recirculación se convierte en un sistema de emergencia de simple circulación por gravedad, cuyo flujo es de 5 m^3 por minuto y se alcanza en 8 segundos. El período de transición entre la recirculación normal y el estado de flujo de emergencia total es de la mayor importancia al analizar los riesgos, ya que durante este período hay que asegurarse de que el flujo y la potencia remanentes se mantienen tales que la refrigeración es suficiente para evitar daños al combustible.

Durante el período de puesta en servicio se realizaron diversos ensayos para determinar el comportamiento del sistema en cuanto al flujo y a la presión mientras se realiza el cambio del sistema normal de recirculación al de circulación simple. Los datos experimentales acerca de la disminución del flujo remanente fueron concordantes en general con las características hidrodinámicas previstas del sistema, aunque se observaron oscilaciones de flujo y presión. Se construyó un programa para la calculadora a fin de obtener las temperaturas de las vainas de aluminio y de la masa de refrigerante (en el período de cambio) a lo largo de las barras combustibles, usando los datos teóricos acerca del flujo remanente y del decrecimiento de potencia. También se programó la máquina para estimar las temperaturas de saturación del refrigerante y detectar cualquier ebullición en estas condiciones. Al superponer sobre los resultados de la calculadora, las oscilaciones de flujo y presión que se observaron en las pruebas de puesta en servicio, resultó una incertidumbre tan grande que parecían posibles situaciones de riesgo de ebullición nucleadas o en masa o ambas a la vez. A resultas de nuevas investigaciones, la potencia del reactor se limitó a 30 MW y la salida por barra a 300 kW.

Se estudiaron diversas modificaciones del sistema de refrigeración para mejorar caudales y presiones durante el período de transición. Se repitieron los ensayos de flujo remanente para comprobar la reproductibilidad de los resultados previos. A la vez se

hicieron medidas experimentales de los transitorios térmicos del refrigerante a la salida de las barras combustibles en condiciones de operación intermedias de la potencia del reactor a fin de comprobar la precisión de las predicciones de la calculadora. Ni en éstos ni en los subsiguientes ensayos de flujo remanente se observaron oscilaciones de presión ni de flujo. Como las temperaturas observadas eran sistemáticamente muy inferiores a los valores predichos, era evidente que no había riesgo de que se fundiese el envainado de aluminio de las barras combustibles o de que se alcanzara la temperatura de transición de fase del uranio. No obstante, al operar a 40 MW las temperaturas observadas indicaban ebullición nucleada. Por tanto se concluyó que para operar el reactor a 40 MW se requerían ciertas modificaciones del sistema primario de refrigeración. Se mejoraron las características hidrodinámicas del sistema instalando volantes en cada una de las bombas de refrigeración primaria y se repitieron los ensayos de amortiguamiento del flujo remanente. Al introducir estos datos en un programa de la calculadora, resultó que no había ebullición de ninguna clase en las barras combustibles para una potencia del reactor en la que la salida por barra combustible no exceda de 330 kW. Para salidas por barra de 375 kW no se observó más que una ligera ebullición nucleada. En nuevas medidas experimentales de los transitorios térmicos del envainado y el agua saliente, se vió que durante el período de sustitución que sigue a la parada de las bombas, la elevación sobre los valores normales de equilibrio propios de barras combustibles sometidas a condiciones de operación con potencia elevada era pequeña o nula.

Puesto que los valores experimentales y el análisis de la calculadora establecían firmemente que no había peligro al operar el reactor con potencia elevada, se le llevó hasta el valor total proyectado de 40 MW y funcionó satisfactoriamente.

Double reactor containment for a circulating liquid fuel reactor and site requirements

By H. Wijker *

The circulating liquid fuel reactor to be considered is the KEMA Suspension Test Reactor in which a suspension of UO_2 and ThO_2 particles in light water is circulated [1]. The KEMA grounds have been chosen as site for this reactor. The nearest inhabited building is the house of a railway employee at 150 m distance (r in Fig. 1), and the next house is at 350 m. Here the population centre U in Fig. 1 starts. A railway line runs as near as 35 m from the centre of the reactor hall.

In terms of site requirements handled by the United States Atomic Energy Commission [2], the boundary of the exclusion area has been chosen at a distance of 35 m from the centre of the hall and the boundary of the low population zone at 150 m. In accordance with this last reference, the population centre distance must thus be at least (4/3) × 150 m = 200 m. This requirement is met in the case under consideration. In the event of a major accident, the total doses at the nearest boundaries to the zones mentioned in a 2 hour period, and an unlimited exposure time, must in all cases be respectively less than a 25 rem total body dose or 300 rem thyroid dose. These requirements have led to the concept of a double reactor containment and adsorption beds for halogens and fission gases.

As this is not the normal containment for a nuclear reactor, a description of this system will be given in this paper.

The release of radioactivity will be considered under the following conditions:

(*a*) Normal operation,

(*b*) Maintenance of the reactor system,

(*c*) A major accident, and

(*d*) A hypothetical accident.

The first two conditions are self-explanatory. A major accident is defined as *an accident in which the total reactor content is released into the inner reactor containment.* A hypothetical accident is an accident with the release of all gaseous fission products and 50% of the halogens into the outer reactor containment.

DESCRIPTION OF THE CONTAINMENT

The reactor and its components are built in steel compartments placed in the reactor hall. There are therefore, three barriers between the fuel and the outside air, *viz.*, the stainless steel of the circuits, the walls of the compartments (thick lines in Fig. 2) and the walls of the reactor hall.

The pressure in the primary circuit where the fuel is circulated, as well as in the gas circuit (Fig. 2) will be about 60 atm at normal operation. The system, however, is designed for 80 atm with a safety margin of at least 60%. It will be tested at 120 atm. The reactor vessel R with walls about 5 mm thick, however, need only stand an inner overpressure of about 35 atm, as it is surrounded by a high pressure vessel PV with walls 5 cm thick, in which CO_2 is circulated under a pressure of 40 atm, which serves as a coolant for the beryllia and graphite reflectors.

The walls of the compartments are 8 mm thick steel plates backed by thick concrete biological shields.

The walls of the reactor hall are concrete, part 50 cm thick, and part 30 cm. On the inside they are painted with an epoxy point, and covered on the outside with a bitumen layer. This containment has two main openings to the outer air during normal operation, namely the inlet duct for the ventilation air and the outlet to the stack.

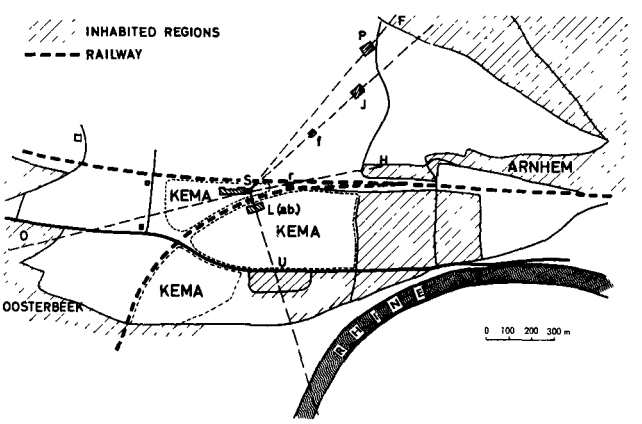


Figure 1. Site of the KEMA Suspension Test Reactor. S = stack

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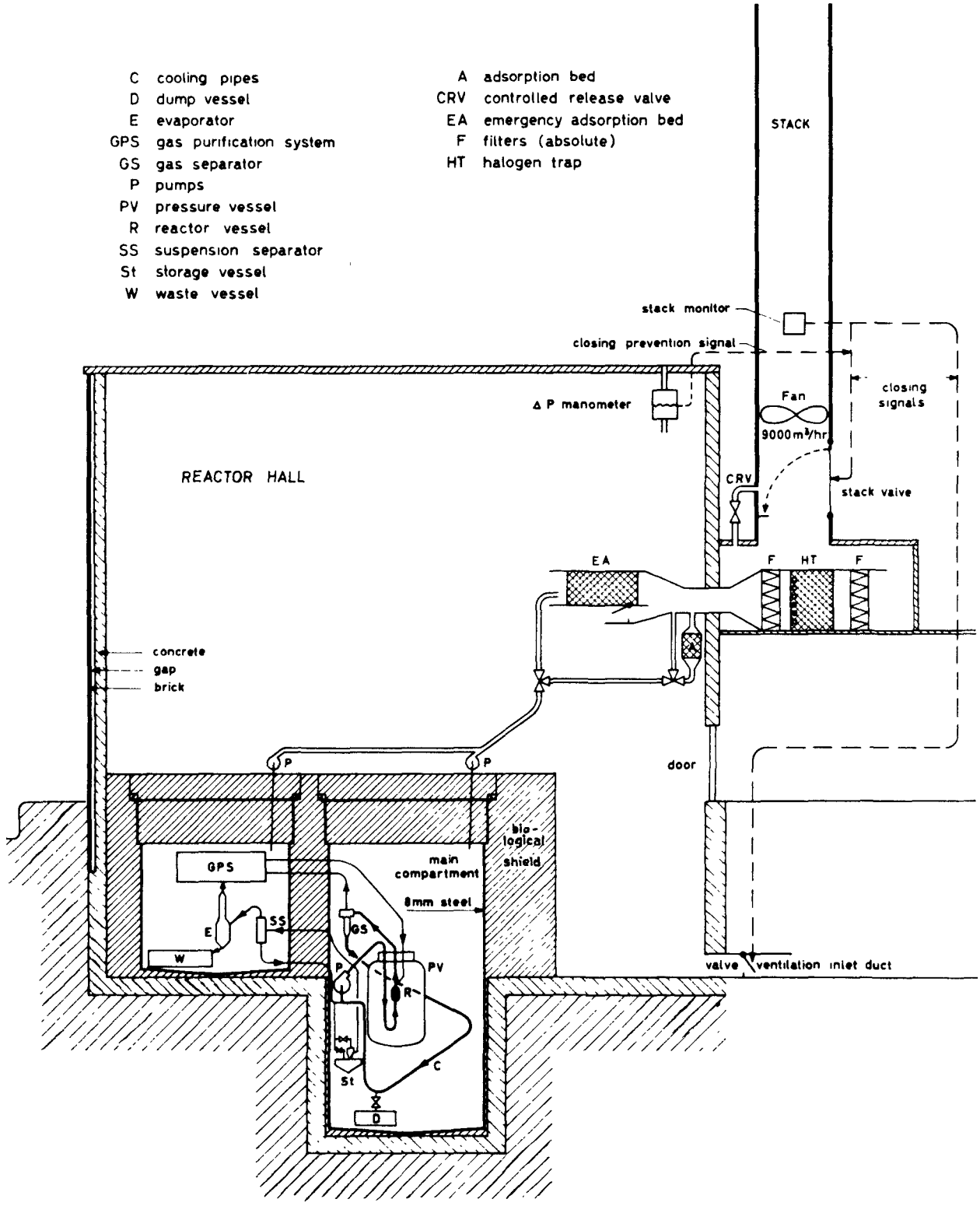


Figure 2. The reactor hall with the waste-gas system and the main and purification compartments containing the primary circuit of the KEMA Suspension Test Reactor

In order to meet thermal requirements, the concrete wall is surrounded by a brick wall. There is a 15 cm gap between the concrete and the brick. Although the latter cannot be considered as a barrier it will delay the leakage of radioactive gases into the outer atmosphere that might have leaked through the concrete wall.

THE LEAKTIGHTNESS OF THE CONTAINMENT

The pressure in the reactor hall will be maintained at 0.005 atm below the barometric pressure. Measurements have shown that, with the ducts, closed the inleakage from the outer air at this pressure difference is 1 % of the hall volume per day. These measurements were carried out with small, thin-walled glass reference bulbs situated at various points in the reactor hall and connected together. These measurements can be repeated at any time required.

Also for the walls of the compartments, there is no requirement for them to be completely leaktight, although all seals are welded. Experience obtained with the homogeneous reactor experiments at Oak Ridge has shown that, in practice, a leakage rate of less than 100 std cm^3/min at a pressure difference of 0.2 atm can be obtained. If a compartment is opened for maintenance purposes, it is questionable whether after closing the same stringent leaktightness requirements can be met. In this case the pressure difference can be decreased to 0.05 atm in order not to exceed the inleakage limit of 100 std cm^3/min.

The compartments are filled with nitrogen to avoid the formation of explosive mixtures with the hydrogen used to pressurize the reactor. These compartments are purged with nitrogen at the rate of 0.25 std m^3/hr. This makes possible a continuous control of the leaktightness by measuring the oxygen content of the purged gas.

The air leak rate of 100 std cm^3/min from the hall into the compartment will give an equilibrium content of ½ volume % oxygen. Concentrations in the region of 0.01 % to 0.1 % are easily measured continuously, so that the required leaktightness can be controlled accurately.

The containment system is designed in such a way that it can cope even with a rupture in the reactor system. The chance that such a major accident will occur is small. Reactor kinetics calculations [3], for instance, have shown that the pressure rise during a fast increase in reactivity is reasonably low, as a result of the prompt negative temperature coefficient.

To give an example: if, in excess of the average fuel concentration, 7 % of the fuel content of the outer system, i.e., that part of the primary circuit outside the reactor vessel, enters the reactor vessel continuously within 3 seconds, the pressure rise is only a few atmospheres in the first 3 seconds after the start of the excursion. If an excess of 20 % enters the vessel, a peak of 40 atm overpressure can arise. In order to avoid a rupture in the reactor vessel, safety valves come into operation when the pressure difference between the primary circuit and the reflector cooling circuit reaches a value of about 30 atm, so that gas can escape to a spare tank.

In our opinion it is incredible that the integrity of the inner reactor containment would be violated by a major accident. Nevertheless, it will be shown that even a hypothetical accident, in which all fission gases and 50 % of the halogens are released into the hall, will not lead to unacceptably dangerous situations.

THE WASTE-GAS SYSTEM

The compartments have to be purged with nitrogen and this gas may contain some radioactivity, especially if the primary system is not completely leaktight. For reactor operation at 250 kW and a leak rate from the gas system of the reactor lower than 0.25 std cm^3/h, the concentration in the stack will not exceed 10^{-7} c/m^3, the upper limit set by the licensing authorities for design purposes. For slightly greater leak rates an adsorption bed A (Fig. 2) is available which gives a delay of about 100 h for xenon and 10 h for krypton. This bed can also be used when slightly contaminated compartments have to be cleaned by purging them with gas before they are opened for maintenance. All gases to be discharged through the stack have to pass the absolute filters F and a halogen trap HT (Fig. 2). A major accident in which all fission products are released into the compartments can cause small overpressures in those compartments in a short time. This time, however, is long enough for the release of so much radioactivity through the compartment walls which are not absolutely leaktight, into the reactor hall, where the concentration would exceed 10^{-4} c/m^3. This is the concentration at which the stack monitor gives signals to close the stack valve and the valve in the ventilation inlet duct. However, closing the stack is automatically accompanied by the opening of an inlet through which 9 000 m^3/h atmospheric air is drawn by the stack fan (Fig. 2). Small amounts of gas can be released from the compartment into this flow via the emergency adsoption bed EA and the controlled release valve CRV, bypassing the stack valve.

In the case of a hypothetical accident in which all fission gases are released into the hall, there may be a small build-up of an overpressure in the hall. The same might happen if the atmospheric pressure outside the hall decreases. In order to prevent this it might be necessary to release a small amount of contaminated air in a short time form the reactor hall. During this time the signal from the stack monitor which serves to close the stack valve, is blocked until the pressure is reduced to a chosen limit. This blocking signal is

given by a manometer (Fig. 2). To maintain an underpressure in the reactor hall, gas has to be removed from it in order to compensate for inleaking air. This gas will have to pass the emergency adsorption bed.

RELEASE OF RADIOACTIVE GASES

Normal operation

During normal operation at 250 kW the reactor contains about 180 kc Kr and Xe isotopes and 100 kc halogens with a half-life greater than 1 second. Equilibrium has been assumed here for all nuclides except ^{85}Kr. For ^{85}Kr the 1 000 MWh value was taken. This ^{85}Kr activity, about 0.005 kc, is negligible for the cases to be considered.

For a gas leakage of v std cm^3/hour from the reactor system into the purification compartment and total mixing, the compartment, which is purged with 0.25 std m^3 nitrogen per hour, will contain, under equilibrium conditions, 1.6 v curie of noble gases, so that the concentration is $1.3 \times 10^{-2}v$ c/m^3. After dilution with 9 000 m^3/h ventilation air the concentration in the stack will be $4 \times 10^{-7}v$ c/m^3. For v smaller than 0.25 cm^3/h this concentration is below the design value. The adsorption bed A (Fig. 2) will reduce the activity with a factor of about 4, so that, when release takes place via this bed, the concentration in the stack is $10^{-7}v$ c/m^3. Therefore, a maximum leakage of 1 std cm^3/h could be accepted.

After replacing some components, which may have to be handled remotely, it is questionable whether the same leaktightness can be obtained. Relaxing requirements by a factor of 10, the concentration in the stack will be 10^{-6} c/m^3. This is nearly 100% ^{133}Xe for which the MPC value for radiological workers, based on a 168 h exposure per week, is 3×10^{-6} c/m^3. Therefore, reactor operation under these conditions can be accepted.

The halogen traps, through which all gases have to pass before discharge, can be considered as impenetrable for iodine and bromine, so that their nuclides can be omitted from consideration.

Maintenance

Before a compartment can be opened for maintenance, radioactive gases have to be removed. It is purged with w std m^3/h. The value w is chosen in such a way that the concentration in the stack exit is lower than 10^{-4} c/m^3. As the compartment to be purged will contain a maximum of 16 c fission gases from normal operation conditions, w must be lower than about 25 std m^3/h.

The concentration A_i of each nuclide in the compartment can be obtained from the equation

$$dA_i/dt = -(\lambda_i + w/V)\,A_i$$

in which λ_i is the decay constant and V the volume of the compartment.

After two days the compartment contains 80% ^{133}Xe, 20% ^{135}Xe and negligible amounts of the other fission gases. The concentration is 10^{-6} c/m^3, which is nearly 1/10 of the maximum permissible concentration of the mixture for an exposure of 40 h/week for radiological workers. As regards fission gases, the compartment can now be opened safely. It can be also shown that iodine nuclides are not a restriction. If the iodine nuclides behave as fission gases, the content in the compartment before flushing would be 0.7 c and the concentration after 2 days purging with 25 m^3/h lower than 4×10^{-7} c/m^3, which is about the acceptable MPC value for the insoluble material.

The major accident

As a consequence of a rupture in the reactor and the cooling circuits, their contents will be released into the compartments. Before the steam condenses, the highest pressure that can be expected in a compartment is 1.56 atm at a temperature of 68 °C. Cooling to 30 °C will reduce the pressure to 1.31 atm.

In order to reduce the leakage of radioactivity from the compartments into the reactor hall, about 100 m^3 will have to be discharged ot the stack within 1 hour. If a clean emergency adsorption bed is used, in which krypton has a delay time of 1 hour, the delay time of xenon is about 10 times as long, no fission gases will pass through the bed from a discharge within this hour. If no condensation has taken place, the pressure in the compartments will have been reduced to 0.98 atm by this procedure. During this time fission gases will escape to the reactor hall. The compartment would contain 180 kc fission gases just after the accident. Assuming a maximum leakage through the containment wall of 100 std cm^3/min at a 0.05 atm pressure difference, 9 curies would reach the reactor hall, giving a concentration of 3×10^{-3} c/m^3. The stack valve and the ventilation inlet duct valve, can close within 1 minute. As only about 40 litres leak from the compartment into the hall, the pressure in the hall will not notably change. After about 3 weeks the concentration in the hall will have decreased by decay to 10^{-4} c/m^3 and ventilation can then be restored.

After 1 month the activity of the fission gases in the compartment will be 250 c. It can be driven away by filling the compartments with about 3 m^3/h of water. Using the adsorption bed A (Fig. 2), the concentration in the stack will be just below 10^{-4} c/m^3.

The hypothetical accident

An accident in which all the radioactive gases escape from the primary loop and which, at the same time, causes a leak in the compartment walls of such a nature that these gases escape into the hall, is in our opinion, incompatible with real possibilities, particularly in view of the results of reactor kinetics calculations. This does

not alter the fact that the precautions that have been taken ensure that a hypothetical accident of this kind can be kept under control as regards to its consequences, in such a way that no hazard results for the environments or the personnel.

In accordance with [2], it is assumed that all fission gases and the half of the halogens enter the hall without delay (180 kc Kr and Xe and 50 kc I and Br). If the stack valve and the inlet valve are closed, the accident would have been accompanied by a pressure rise of up to 1.10 atm, assuming the escape of all the water from the primary cooling system at the same time and without condensing at all. The blocking signal, however, (Fig. 2) prevents the closure of the stack valve unless the pressure is lower than 1 005 atm. Thus 9.5% of the activity content of the reactor hall is blown to the stack. Releasing with a rate of 9 000 m³/h, this process takes about 2 minutes, during which time 10 kc of fission gases escape. If the halogen traps have, say, at this release rate, a decontamination factor of only 10^4, instead of being impenetrable, 3 curie of halogens will escape. The closure of the stack valve has to be completed within 18 s. In 18 s the maximum underpressure premissible for the hall, 0.01 atm, will have been reached. If the closure takes 18 s, 13 kc of fission gases will have been discharged from the stack. If the valve closes in a negligible time, gas will be released through the controlled release valve (CRV in Fig. 2) until an underpressure of 0.005 atm is reached. With a release rate of 5 m³/h this operation takes 6 minutes.

During the time that the hall has an overpressure in relation to the outer atmosphere, fission gases will leak through the second reactor containment, namely, less than 18 c in the first 2 minutes and further 0.8 c during a controlled release in the case of an immediate closure of the stack valve. For halogens the values are 6 c and 0.3 c.

If an underpressure of 0.005 atm is obtained, it is maintained by a controlled release of about 1 m³/h, in order to compensate for air infiltration from the outside. That means a discharge rate of about 25 c/h of fission gases, which decreases as a consequence of the decay of the gases in the hall. Taking into account this decay, the integrated activity released by this process is 270 c.

After 1 month the hall will contain 230 c of fission gases, and after 3 months they will have decayed to 5 c. Under favourable meteorological conditions this residual activity can be vented to atmosphere.

ATMOSPHERIC DILUTION AFTER A DISCHARGE FROM THE STACK

Calculations of the dilution of exhaust gases, before reaching points of interest, have been based on the well known formula of Sutton:

$$C = (Q/\pi u c^2 x^{2-n}) \cdot \left(e^{-(h-z)^2/c^2 x^{2-n}} + e^{-(h+z)^2/c^2 x^{2-n}}\right)$$

in which C is the concentration in curie/m³ at a distance of x meters from the stack in the wind direction and at z meters above ground level, if the discharge is Q curie/s. Furthermore u is the wind speed in m/s, h the effective height of he stack in meters and n and c^2 the turbulence and diffusion parameters.

This concentration has a ground level maximum of $C_{mo} = 2\,Q/\pi u e h^2$ (e is the base of natural logarithms) for $x = x_{mo}$ given by the formula $h^2/c^2 x_{mo}^{2-n} = 1$.

As the points of interest are at a short distance, it appeared necessary to study the concentrations in the plane through the stack and the wind direction and not only at ground level. To simplify this, new variables are introduced, namely

$$X = (x/x_{mo})^{1-\frac{1}{2}n} = (c/h)\, x^{1-\frac{1}{2}n}$$
$$Z = z/h$$
$$C_r = = C/C_{mo}$$

In these dimensionless variables, Sutton's equation will be:

$$C_r = (\tfrac{1}{2}e/X^2)\left(e^{-(1-Z)^2/X^2} + e^{-(1+Z)^2/X^2}\right).$$

Figure 3 gives lines of constant relative concentrations in the X Z plane. Lines of relative maxima in the X and Z directions are indicated as well as the signs of the partial derivates of C_r in the regions enclosed by these lines and the axes. For X greater than $\sqrt{2}$, the concentration always decreases with height and the usual method to consider only ground concentrations can be applied here as it is on the safe side.

The line $\delta C_r/\delta Z = 0$ can be considered as the axis of the plume. The figure shows that this axis hits the ground at $X = \sqrt{2}$.

In order to find the points of maximum concentration in a block of houses, the value of X has to be derived from the distance x of the nearest house. This value depends on the weather condition to be considered. Calculations have been carried out for the four weather conditions introduced by Sutton [4]. These are indicated by the indices s for a stable condition (moderate inversion), i for instability (zero small temperature gradient), s_+ for very stable (large inversion) and i_+ for very unstable (large lapse rate). For three of these conditions, the relative positions of the block of houses H in Fig. 1, at a 500 m minimum distance from the stack, are given in Fig. 3. The points of maximum concentration, P, Q and R in the figure, are at the top of the houses. For the highly unstable condition i_+ the X value appeared to be 2.4, so that the point of highest concentration is at ground level at the site of the house nearest to the stack. For the flats F at a distance of 1 km from the stack, the point "T" of highest concentration in the s_+ condition is lower than the roof level, although near to it.

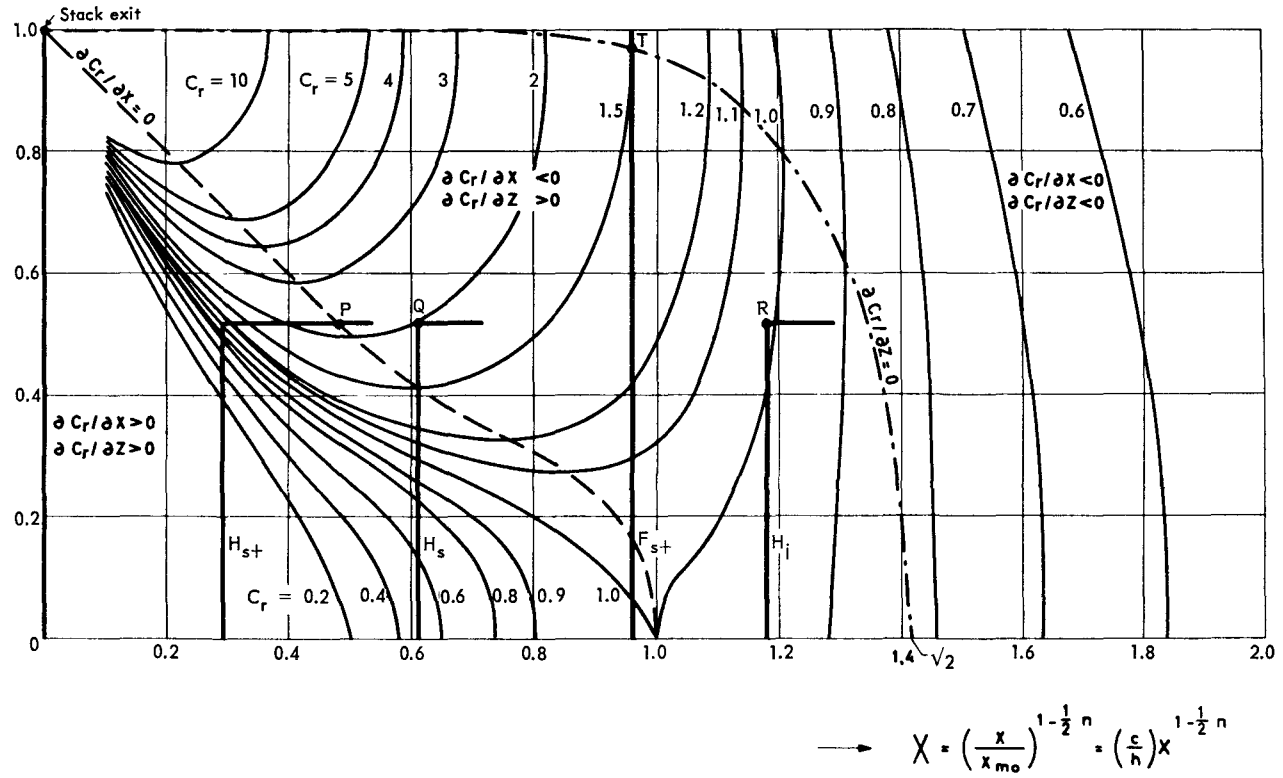


Figure 3. Lines of constant relative concentration C_r and of relative maxima in X-direction (dashed line) and in Z direction (dash dotted line) in the plane through the stack and wind direction. The lines H and F give the relative positions of the houses H and flats F in Figure 1 for unstable (i), stable (s) and very stable (s_+) weather conditions

After the points of maximum concentration had been found, the atmospheric dilution factor $a = C/C_s$ was calculated. Here C_s is the concentration at the stack exit, so that for a discharge of 9 000 m³/h, the source Q in the Sutton formula is $Q = (9\,000/3\,600)\,C_s$.

All values found were smaller than, and in general, appreciably smaller than that found for the point T in Fig. 3, for which $a = 6 \times 10^{-3}/u$. As the lowest wind speed to be considered is about 1 m/sec, it is on the safe side to use an atmospheric dilution factor of 0.01.

A CHECK WHETHER KSTR MEETS SITE REQUIREMENTS

The first site requirement, according to [2] concerns the total body dose for the population at the zone boundaries. The dose caused by a passing radioactive cloud has to be considered only in the cases of major and the hypothetical accidents.

In the case of a major accident, the procedure described above includes a discharge of 250 curie of ^{133}Xe. This gives a dose of 55 mrem as a maximum.

Moreover, discharge can be delayed until meteorological conditions are favourable and the wind is in the direction of the woods so that the exposure of the public is nearly nil.

As practically all other fission products that can escape from the reactor circuit stay inside the biological shield, the dose to the surroundings from this contained radioactivity can be neglected. So in this case, KSTR meets the first site requirement.

In the case of a hypothetical accident, an uncontrolled release of about 13 kc of fission gases may occur. Therefore, the integrated dose from the passing cloud may be as high as 9 rem, but is only a few per cent of a rem at distances of 35 m and 150 m from the stack. A further release of 270 + 230 = 500 c (see above) can be made for the most part under suitable weather conditions and will never give a dose more of than about 0.3 rem. The contents of the hall, 180 kc of fission gases and 50 kc of halogens, give at 35 m a dose of 7 rem in 2 h from the fission gases and 18 rem in 2 h from the halogens, so that the total dose is within the limits required. At the boundary of the low population zone, where the nearest house is situated and which is at a distance of 150 m from the stack, the total external dose for an infinite time is 1 rem from the fission gases and 5 rem from the halogens. Thus, in this case the total dose, 6 rem, is within the required limits as well.

The second site requirement, namely that the thyroid dose be smaller than 300 rem, is met because, thanks to the halogen trap, no iodine and bromine is released

to the environs as a rule. A contingent release of 3 c would give a thyroid dose smaller than about 1 rem. Only within the exclusion area would a small part of the 6 c of halogens leaking out of the reactor hall reach the outer atmosphere together with a small amount of fission gases. Here the delay due to hold-up may play an important part. Without this delay the thyroid dose at 12 m from the hall (the nearest distance to a road) will be lower than about 60 rem in two hours. This amount is low enough to warrant a safe evacuation of the personnel from the site.

Therefore, the conclusion can be drawn that the KEMA Slurry Test Reactor satisfies reasonable safety requirements, and more specifically, it meets US AEC site requirements.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/800 Pays-Bas

Enceinte double d'une pile à combustible liquide circulant et conditions imposées au site

par H. Wijker

Le « KEMA Suspension Test Reactor » de 250 kW, pile à combustible liquide circulant, est en construction sur les terrains de KEMA près d'Arnhem, aux Pays-Bas. Le principe de l'enceinte double du réacteur a été appliqué pour répondre aux conditions en matière d'implantation relatives aux doses limites aux confins des zones, telles qu'elles sont appliquées par l'USAEC.

Le système du réacteur (première barrière entre la suspension et l'atmosphère extérieure) est placé dans des compartiments d'acier; ceux-ci sont doublés par le béton de l'écran biologique (deuxième barrière, première enceinte du réacteur), l'écran à son tour est placé dans un hall de béton (troisième barrière, seconde enceinte du réacteur).

Quatre cas dans lesquels un dégagement accidentel de gaz radioactif peut se produire seront considérés; notamment lors du fonctionnement normal, de l'entretien, d'un accident grave et d'un accident hypothétique.

La concentration maximale de la radioactivité dans les gaz évacués par la cheminée (hauteur 25 m) a été fixée par les autorités à 10^{-7} curie/m^3. Pour maintenir la sous-pression dans les compartiments, qui sont remplis d'azote, pendant le fonctionnement normal, les gaz traverseront des lits d'adsorption à charbon avant d'être évacués par la cheminée. De plus, les gaz passeront par des filtres absolus et des capteurs d'iode.

Pour l'entretien après l'arrêt du réacteur, les compartiments doivent être ouverts. Auparavant, il faut que les compartiments soient pratiquement vidés des gaz radioactifs qui peuvent s'être échappés du réacteur. Cette condition est remplie par l'effet de la décroissance radioactive combinée avec l'épuration. Afin de réaliser cette condition dans un délai raisonnable, la concentration de la radioactivité dans l'effluent de la cheminée doit excéder 10^{-7} curie/m^3 pendant de courtes périodes. Cette concentration n'atteindra pas la valeur de 10^{-4} curie/m^3, soit la limite qui déclenche la fermeture automatique des soupapes de la cheminée et du conduit d'entrée de la ventilation.

En cas d'accident grave (dégagement du contenu du réacteur dans les compartiments étanches), cette limite est atteinte et les soupapes se ferment. L'argumentation selon laquelle l'intégrité de la première enceinte reste intacte est discutée. La décroissance et l'évacuation contrôlée par l'intermédiaire d'un lit d'adsorption à charbon permettent d'expulser pratiquement tous les gaz radioactifs des compartiments en l'espace de trois mois sans violer les conditions imposées au site.

Ces conditions peuvent être respectées même dans le cas hypothétique d'un accident au cours duquel tous les produits de fission gazeux (700 curies) sont libérés dans le hall du réacteur. Pour le démontrer il faut considérer la dilution météorologique qui a lieu entre la cheminée et le récepteur de doses.

A cause de la proximité des régions habitées, la concentration radioactive au-dessus du niveau du sol doit aussi être prise en considération. Pour localiser l'endroit de la concentration maximale dans les régions habitées, un graphique (utilisant la formule de Sutton) a été tracé pour un plan vertical déterminé par la cheminée et la direction du vent.

A/800 Нидерланды

Создание двойного защитного корпуса реактора с циркулирующим жидким ядерным топливом и требования, связанные с выбором места строительства

Х. Вийкер

В настоящее время на площадках фирмы КЕМА вблизи Арнхема, Нидерланды, ведется строительство суспензионного (с циркулирующим жидким ядерным топливом) реактора мощностью 250 *квт*. Использование принципа двойного защитного ограждения реактора связано с необходимостью удовлетворения местных требований в отношении предельных значений доз на границах зоны отчуждения, как это имеет место в практике Комиссии по атомной энергии США.

Реакторная система (первый барьер между суспензионным ядерным топливом и внешней атмосферой) располагается в стальных камерах, окруженных бетонной биологической защитой (второй барьер, первая защитная оболочка реактора), которая, в свою очередь, находится внутри бетонного реакторного зала (третий барьер, вторая защитная оболочка реактора).

Возможное высвобождение радиоактивных газов в окружающую среду рассматривается в докладе для четырех случаев: 1) нормальной работы; 2) обслуживания и ремонта; 3) большой аварии и 4) гипотетической аварии.

Официальными руководящими органами установлено для проектных целей, что максимальная концентрация радиоактивности в газах, выбрасываемых из вытяжной трубы высотой 25 *м*, должна составлять 10^{-7} *кюри/м*^{3}. Для поддержания во время нормальной работы установки пониженного давления в камерах, заполненных азотом, газы удаляются по вытяжной трубе через слои древесного угля для замедления потока и всегда пропускаются через абсолютные фильтры и ловушки, улавливающие иод.

Для проведения осмотра и ремонтных работ после остановки реактора необходимо открыть доступ в камеры. Прежде всего из них практически должны быть удалены радиоактивные газы, которые могли попасть туда из реакторной системы. Это достигается в результате распада радиоактивных веществ и продувки камер. В целях проведения этих операций в приемлемый отрезок времени концентрация радиоактивности в выбросах из вытяжной трубы должна превышать 10^{-7} *кюри/м*^{3} на короткий период времени. Эта концентрация не достигнет уровня, равного 10^{-4} *кюри/м*^{3}, который является предельным и при котором автоматически закрываются клапаны в вытяжной трубе и во входной вентиляционной трубе.

В случае крупной аварии (попадание радиоактивности из реактора в соседние неповрежденные камеры) достигается вышеуказанный предельный уровень и срабатывают клапаны. Обсуждается проблема сохранения целостности первой защитной оболочки. Снижение активности в результате распада и контролируемое удаление активности через аварийный слой древесного угля позволяют вывести практически все радиоактивные газы из камер приблизительно через 3 *мин*, не нарушая местных требований безопасности.

Даже при гипотетической аварии, при которой все газообразные продукты деления (700 *кюри*) попадают в реакторный зал, можно добиться соблюдения местных требований безопасности.

В докладе указывается на необходимость рассмотрения вопроса о метеорологическом разбавлении между вытяжной трубой и дозиметром. Ввиду небольшой удаленности установки от населенных районов необходимо исследовать также степень концентрации радиоактивности над уровнем земли. В целях определения точки максимальной концентрации радиоактивности в данных населенных районах был построен график, основанный на формуле Саттона, для вертикальной плоскости через вытяжную трубу и направление ветра.

A/800 Paises Bajos

El recubrimiento doble de un reactor para un reactor de combustible líquido circulante y los requisitos de emplazamiento

por H. Wijker

Se está construyendo en los campos de KEMA, cerca de Arnhem (Países Bajos) el reactor de ensayo de suspensión KEMA, de 250 kW, que es un reactor de combustible líquido circulante. Para cumplir con los requisitos de emplazamiento referentes a las dosis límites en los bordes de la zona, tal como lo emplea la USAEC se usa el principio de recubrimiento doble del reactor.

El reactor está contenido en compartimientos de acero (primera barrera entre la suspensión y la atmósfera exterior) apoyados en el hormigón del blindaje biológico (segunda barrera, primer recubrimiento) que a su vez está emplazado en la nave del reactor que es de hormigón (tercera barrera, segundo recubrimiento del reactor).

Se considera la posibilidad de escape de gases radiactivos hacia el ambiante exterior para 4 casos, a saber: operación normal, entretenimiento, un accidente grave y un accidente hipotético.

Las autoridades establecen que en el proyecto la concentración máxima de radiactividad en los gases liberados por la chimenea de 25 m de altura ha de ser de 10^{-7} curios/m^3. A fin de mantener la baja presión en los compartimientos llenos de nitrógeno, durante la operación, se descargan los gases a través de la chimenea pasando por capas de carbón que los retardan y siempre pasan por filtros absolutos y trampas de yodo. El entretenimiento obliga a abrir los compartimientos después de parar el reactor. Antes deben estar prácticamente libres de gases radiactivos que pudieran haber escapado del sistema del reactor. Esto se consigue combinando el decrecimiento natural con purgas. Para que esto se consiga en un tiempo razonable la concentración de radiactividad en el gas que fluye por la chimenea tiene que exceder de 10^{-7} curios/m^3 por breves intervalos. Esta concentración no alcanzará el valor de 10^{-4} curios/m^3 porque a este límite las válvulas de la chimenea y el conducto de entrada de la ventilación se cerrarán automáticamente.

En el caso de un accidente grave (liberación del contenido del reactor hacia los compartimientos estancos) se alcanza este límite y las válvulas se cierran. Se discuten las zonas por las que se mantiene intacta la integridad del primer recubrimiento. Por decrecimiento natural y mediante la liberación controlada pasando por una capa absorbente de carbón de emergencia se puede extraer prácticamente todo gas radiactivo de los compartimentos en unos 3 meses sin violar los requisitos de emplazamiento.

Incluso en el caso de un accidente hipotético en el que todos los gases de fisión (700 kcurios) se liberasen en la nave del reactor, se pueden cumplir los requisitos de emplazamiento.

Para demostrar esto hay que tener en cuenta entre otros puntos la dilución meteorológica entre la chimenea y el receptor de la dosis. También hay que considerar la concentración de radiactividad en niveles altos, a causa de la cercanía de regiones habitadas. Basándose en la fórmula de Sutton se ha construído un gráfico, para un plano vertical que contenga la chimenea y la dirección del viento, que permite localizar el punto de máxima concentración en estas regiones habitadas.

Вопросы безопасности атомных судов

В. И. Неганов, Б. Г. Пологих, Ю. В. Сивинцев

В настоящее время накоплен опыт эксплуатации первых гражданских судовых ядерных энергетических установок (ЯЭУ), в частности на ледоколе «Ленин». В связи с этим представляется важным проанализировать полученные данные с точки зрения безопасности атомных судов и тем самым сделать выводы о правильности или недостатках принятых ранее решений и мероприятий. С учетом этих соображений ниже рассмотрены некоторые вопросы безопасности эксплуатации ледокола «Ленин».

Ядерная энергетическая установка ледокола описана в работах [1-3]. Опыт ее эксплуатации в 1959—1963 гг. изложен в докладе для Третьей Женевской конференции [4].

СУДОВАЯ ЯЭУ ЛЕДОКОЛА — НАДЕЖНЫЙ ИСТОЧНИК ЭНЕРГИИ

Как известно, ЯЭУ представляют существенную потенциальную опасность в связи с наличием большого количества радиоактивных веществ внутри реакторов и их первичных контуров. Авария ЯЭУ с нарушением плотности первичного контура и тем более с нарушением герметичности оболочек тепловыделяющих элементов может привести к распространению (за пределы специально ограниченных зон) радиоактивных веществ в количествах, опасных для персонала, обслуживающего установку, или даже для населения, находящегося вблизи места ее базирования. Это особо важно учитывать при сооружении транспортных ЯЭУ, в частности судовых, о которых будет идти речь.

При решении вопроса об использовании ЯЭУ на судне того или иного назначения должна быть исследована возможность обеспечения повышенной надежности такого судна как своего рода «платформы» для размещения потенциально опасной ЯЭУ.

Это условие вызывает необходимость анализа статистических данных по аварийным ситуациям с морскими судами как в части среднего числа этих событий, так и причин и мест их свершения.

Прежде всего следует обратить внимание на количество аварий, связанных с навалами и столкновениями судов, посадками их на мель, с крупными повреждениями машин и механизмов, с пожарами и взрывами, с поломками в итоге стихийных бедствий (штормов, бурь) и прочими причинами. Соответствующие сведения представлены в табл. 1. Из них следует, что ежегодно практически каждое третье судно с валовой вместимостью более 500 регистровых брутто-тонн (БРТ) попадает в аварию.

Таблица 1. Аварийность на мировом торговом флоте в 1948—1963 гг. (по данным Ливерпульской ассоциации страховщиков [5] для судов с валовой вместимостью более 500 БРТ)

Год	Численность мирового торгового флота	Число судов, потерпевших аварию	Доля мирового торгового флота, %
1948	16 340	7 729	42,3
1949	17 227	7 063	41,0
1950	18 052	6 873	38,1
1951	18 426	7 789	42,3
1952	18 661	7 776	41,7
1953	18 997	7 269	38,3
1954	19 658	7 014	35,6
1955	19 743	—	—
1956	20 289	—	—
1957	20 926	7 333	35,1
1958	21 924	6 944	31,7
1959	22 531	7 359	32,6
1960	22 621	7 368	32,6
1961	23 892	7 818	32,7
1962	24 243	7 938	32,7
1963	24 603	8 008	32,5

Безусловно с точки зрения сохранения целостности ЯЭУ на судне наиболее опасными являются аварийные случаи, связанные со столкновениями судов, пожарами и взрывами, а также с бедствиями во время штормов и бурь. Данные табл. 2, в которой проанализированы причины аварий, свидетельствуют, что количество таких событий является весьма значительным. Так, например, в 1963 г. из 8 тыс. судов, потерпевших аварию, 1814 (22,6%) получили повреждения в столкновениях; 1615 (20,2%) — в навалах; в 1398 случаях (17,5%) имели место повреждения машин, механизмов, валов и винтов; 1049 судов (13,1%) натолкнулись на подводные препятствия; 792 судна (9,9%) были повреждены в штормах и бурях; на 439 судах (5,5%) произошли пожары и взрывы и более 900 судов (11,2%) потерпело аварии по другим причинам (в том числе пропало без вести и затонуло). Табл. 2 содержит

Таблица 2. Количество судов, потерпевших аварии по различным причинам в 1959—1963 гг. (по данным Ливерпульской ассоциации страховщиков[6] для судов с валовой вместимостью более 500 БРТ)

Причина аварии	1959 г.		1960 г.		1961 г.		1962 г.		1963 г.	
	Число судов	%	Число судов	%	Число судов	%	Число судов	%	Число судов	%
Навалы	1683	22,9	1675	22,7	1753	22,4	1594	20,1	1615	20,2
Столкновения	1600*	21,7	1500*	20,4	1600*	20,5	1818	22,9	1814	22,6
Повреждения машин	1229	16,7	1337	18,1	1401	17,9	1362	17,2	1398	17,5
Посадки на мель	989	13,4	1023	13,9	954	12,2	993	12,5	1049	13,1
Штормы, бури	772	10,5	739	10,0	844	10,8	826	10,4	792	9,9
Пожары, взрывы	428	5,8	419	5,7	470	6,0	454	5,7	439	5,5
Прочие	658	9,0	675	9,2	796	10,2	891	11,2	901	11,2
Всего	7359	100	7368	100	7818	100	7938	100	8008	100

* Приблизительно. В сумме с навалами точно.

данные за 1959—1963 гг. Статистические данные по аварийности на мировом флоте в предшествующие годы (1948—1958 гг.) аналогичны приведенным.

Сведения о потерях мирового торгового флота в 1948—1962 гг. (по данным Регистра Ллойда [7] для судов с валовой вместимостью более 100 БРТ) приведены в табл. 3. Следует отметить, что в упомянутую статистику не включены надводные и подводные военные суда. Как видно из этой таблицы, ежегодно погибает 160—250 судов вместимостью более 100 БРТ, что составляет 0,5—0,9% всей численности мирового торгового флота и 0,25—0,35% мирового тоннажа. Данные табл. 4 иллюстрируют тот факт, что гибель является событием достаточно большой вероятности даже для крупных судов с большим водоизмещением, которое должно быть более свойственно судам с ЯЭУ.

Таблица 3. Потери мирового торгового флота в 1948—1962 гг. (по данным Регистра Ллойда [7] для судов с валовой вместимостью более 100 БРТ)

Год	Число погибших судов	Доля численности мирового флота, %	Вместимость погибших судов, тыс. БРТ	Доля мирового тоннажа, %
1948	196	0,67	222,5	0,27
1949	220	0,94	244,1	0,30
1950	222	0,72	260,0	0,31
1951	218	0,70	290,9	0,33
1952	188	0,60	250,2	0,28
1953	226	0,71	322,2	0,35
1954	182	0,56	262,4	0,27
1955	178	0,55	254,7	0,25
1956	163	0,49	248,5	0,24
1957	163	0,48	270,9	0,25
1958	160	0,46	347,5	0,29
1959	181	0,50	281,5	0,23
1960	171	0,47	358,2	0,28
1961	189	0,50	471,1	0,35
1962	249	0,64	481,1	0,34

Весьма интересный материал содержится в табл. 5, где указаны причины гибели судов по данным Регистра Ллойда [8]. Как видно, в 1958—1962 гг. преимущественными причинами гибели являются посадки на подводные препятствия (40—50%), затопление (25—30%), столкновения (10—13%) и пожары и взрывы (10—20%). Аналогичные цифровые данные были получены из таблиц Регистра Ллойда и для предыдущих 1948—1957 гг.

Весьма важно то обстоятельство, что, несмотря на ежегодно увеличивающееся оснащение судов современными навигационными приборами и на возрастающее использование береговых установок для предупреждения судов об опасностях, количество случаев с гибелью судов практически держится на постоянном уровне [9].

Затопление судна с ЯЭУ обусловливает особо большую опасность, так как оно влечет за собой возможное загрязнение донных слоев моря долгоживущими радиоактивными веществами, проникновение их в растительный и животный мир моря и через пищевые цепочки в организмы людей.

Авария судна с ЯЭУ вдали от населенных пунктов и путей интенсивного судоходства безусловно носит характер относительно меньшей опасности. К сожалению, исследования показывают, что аварийные ситуации происходят главным образом вблизи портов. Так, опубликованный в литературе анализ случаев столкновений и гибели морских судов в 1955—1961 гг. показал, что наиболее вероятными местами таких событий являются подступы к порту и рейды (66% всех случаев). Только 8% столкновений судов за эти годы произошли в открытом море [10].

Все изложенное заставляет проявлять максимум осторожности при выборе типа судна для использования на нем ЯЭУ, помня, что в случае затопления такого судна в районе порта или узкого пролива будет необходимо на неопределенный промежуток времени закрыть соответствующий торговый порт или

Таблица 4. Распределение по тоннажу судов, потерпевших аварии в 1961 и 1962 гг.

Вместимость, БРТ	Год	500—4999	5000—9999	более 10 000
Затонувшие суда	1960	12	3	2
	1961	13	4	—
Аварии, связанные с пожарами, взрывами, столкновениями и посадками на мель	1960	54	10	6
	1961	33	28	11
Всего:	1960	66	13	8
	1961	46	32	11

Таблица 5. Количество торговых судов, погибших по различным причинам в 1958—1962 гг. (по данным Регистра Ллойда [8] для судов вместимостью более 100 БРТ)

Причины гибели	1958 г.		1959 г.		1960 г.		1961 г.		1962 г.	
	Число судов	%	Число судов	%	Число судов	%	Число судов	%	Число судов	%
Посадки на мель	56	35,0	73	40,3	72	42,2	80	42,3	123	49,5
Затопление	48	30,0	60	33,1	50	29,2	43	22,8	64	25,7
Пожары, взрывы	15	9,4	14	7,7	17	9,9	35	18,5	24	9,6
Столкновения	23	14,4	22	12,2	19	11,1	23	12,2	16	6,4
Пропали без вести	2	1,2	3	1,7	7	4,1	1	0,5	11	4,4
Прочие	16	10,0	9	5,0	6	3,5	7	3,7	11	4,4
Всего	160	100	181	100	171	100	189	100	249	100

пролив. Наносимый этим экономический ущерб нами не оценивался, но, очевидно, он будет весьма существенным.

В свете изложенного становится ясным принятое в свое время в Советском Союзе решение — на 1 этапе использования, до накопления достаточного опыта эксплуатации гражданских судовых ЯЭУ, применять их на ледоколах, вероятность затопления которых исключительно мала. Характерно, что согласно мировой статистике среди более чем 2000 судов, погибших за последние 15 лет, нет ни одного арктического ледокола.

Таким образом, ледокол является наиболее надежной «платформой», обеспечивающей живучесть энергетической установки в любых эксплуатационных условиях.

Как ранее отмечалось [1], корпус ледокола «Ленин» достаточно прочен для выдерживания любого сжатия льдов, возможного в Арктике. Он разделен 11 главными поперечными водонепроницаемыми переборками на 12 отсеков. При этом обеспечена непотопляемость ледокола при одновременном затоплении даже двух смежных отсеков.

Благодаря наличию особо прочного корпуса ледоколу не страшны обычные морские опасности — столкновения с другим судном, сжатия во льдах, посадка на мель.

Небезынтересно в рассматриваемом аспекте сослаться на следующий случай из эксплуатации ледокола «Ленин».

12 октября 1961 г. ледокол шел в проливе Санникова 17 узловым ходом в мелко битых тонких льдах и натолкнулся на подводный айсберг, осевший на дно. Сопротивление льда оказалось настолько большим, что в результате столкновения ледокол полностью потерял скорость и застрял на подводном препятствии, накренившись на правый борт. После того как ледокол за счет работы собственными винтами сошел со льда, с левого борта на поверхность воды всплыли большие массы обломков льда, а бортовой отсек (между поперечными переборками 36 и 48 шпангоутов) оказался затопленным. Внутренние помещения судна оказались неповрежденными, поскольку практически по всей длине ледокола предусмотрен так называемый второй борт из продольных водонепроницаемых переборок. В районе расположения ЯЭУ они отстоят от наружной обшивки на расстоянии, превышающем 1/5 ширины судна. Опасность получения большого крена при затоплении бортовых отсеков на ледоколе исключена наличием перетоков между ними, обусловливающих при повреждении наружной обшивки затопление отсеков другого борта, симметричное относительно диаметральной плоскости судна.

Несмотря на повреждение, ледокол продолжал нормальную работу в Арктике еще в течение полутора месяцев после столкновения. Осмотр судна в доке показал, что наружная обшивка имела вмятину (на протяжении почти

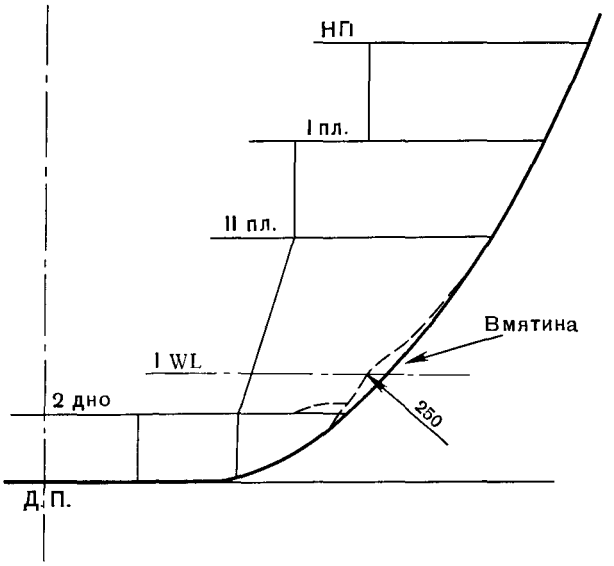


Рис. 1. Повреждение корпуса ледокола «Ленин» при столкновении с подводным ледообразованием в 1961 г. (разрез по 41 шпангоуту, вид в нос).

НП — нижняя палуба; ДП — диаметральная плоскость; I пл.— первая платформа; II пл.— вторая платформа

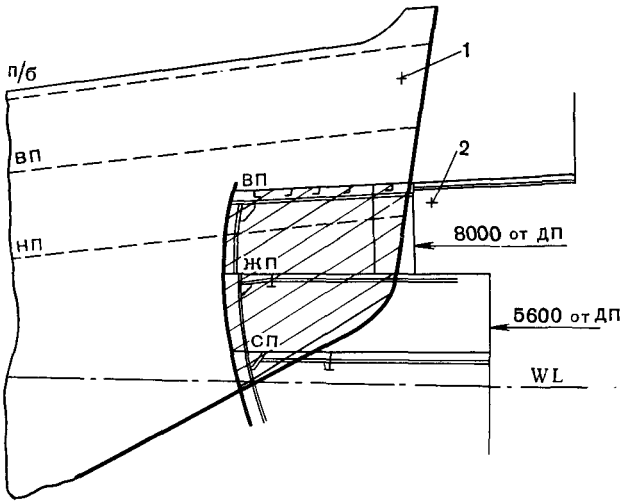


Рис. 2. Схема к расчету гипотетического случая столкновения ледокольного транспорта «Лена» с ледоколом «Ленин». Заштрихована разрушаемая часть борта атомного ледокола «Ленин»:

1 — носовая часть транспорта «Лена»; *2* — борт ледокола «Ленин»; п/б — полубак; ВП — верхняя палуба; ЖП — жилая палуба; СП — средняя палуба; НП —нижняя палуба; ДП — диаметральная плоскость

всего балластного отсека) с наибольшей стрелой прогиба 250 *мм* и разрывами в этом месте (рис. 1). Таким образом, происшедшее столкновение подтвердило высокую прочность корпуса и оптимальное с точки зрения непотопляемости распределение водонепроницаемых отсеков, предусмотренное при проектировании ледокола «Ленин».

Столкновение ледокола с другими судами вряд ли приведет к гибели ледокола. Была произведена расчетная оценка последствий максимально опасного случая для ледокола — столкновения ледокольного транспорта «Лена», обладающего весьма прочным корпусом и скоростью хода 15 узлов при водоизмещении около 11 тыс. *т* с ледоколом «Ленин» при нанесении удара ледоколу под прямым углом к борту напротив места размещения ЯЭУ.

Оказалось, что даже при нанесении удара транспортом «Лена» с полного хода повреждение борта ледокола не распространяется на отсек, в котором расположено радиационно опасное оборудование (рис. 2).

Изложенное подтверждает достаточность мер, предусмотренных для сохранения ледокола «Ленин» на плаву и защиты ЯЭУ от разрушения при возможных случаях навигационных повреждений.

Как во время описанного столкновения ледокола с неподвижным льдом, так и в условиях качки, длительных вибрационных воздействий и непрерывных сотрясений от ударов корпуса ледокола об льды ЯЭУ работала без остановок.

В целях обеспечения безопасности ЯЭУ на ледоколе предусмотрено автоматическое прекращение цепной реакции в активной зоне реактора при недопустимом отклонении одного или нескольких особо важных параметров от заданных значений. Как показал опыт эксплуатации ЯЭУ ледокола «Ленин» (табл. 6), такие события на судовых реакторах не исключены.

Удовлетворительные результаты длительной эксплуатации ЯЭУ ледокола «Ленин» в сложных по непрерывным механическим воздействиям условиях дают основание считать, что ядерные реакторы на судне могут быть надежным источником энергии.

БЕЗОПАСНОСТЬ АТОМНОГО ЛЕДОКОЛА «ЛЕНИН» ДЛЯ ОКРУЖАЮЩЕЙ СРЕДЫ

После вступления в состав ледокольного флота в конце 1959 г. ледокол «Ленин» принимал деятельное участие в проводке судов во льдах Арктики в навигациях 1960—1963 гг. За это время им пройдено около 60 000 *миль*, из них около 40 000 *миль* во льдах. По трассе Северного морского пути с участием ледокола «Ленин» проведено 300 судов.

ЯЭУ ледокола «Ленин» оказалась радиационно безопасной как для обслуживающего персонала, так и для внешней среды. Отметим, что внутри тепловыделяющих элементов каждо-

Таблица 6. Случаи автоматического сброса стержней аварийной защиты (АЗ) во время работы ледокола «Ленин» в Арктике

Номер реактора	1	2	3
Сбросы мощности имели место после выработки	32,8; 36,2; 46,0; 88,9; 115,2 эфф. суток * в 1 кампании и 14,3 эфф. суток во 2 кампании	27,6; 32,6 и 45,3 эфф. суток в 1 кампании и 15,7; 26,1 и 86,3 эфф. суток во 2 кампании	17,0; 17,6; 17,8; 45,0; ,83,2; 83,3 и 195 эфф. суток в 1 кампании и 18,7 эфф. суток во 2 кампании
Суммарное количество сбросов АЗ	7 раз	6 раз	8 раз

* Эффективные сутки соответствуют выработке 90 *Мвт·сутки*.

го реактора ледокола содержатся радиоактивные вещества, суммарная активность которых достигает $3 \cdot 10^8$ *кюри*. Около 10^5 *кюри* активных веществ находится вне материала таблеток тепловыделяющих элементов из спеченной двуокиси урана. После нарушения целостности оболочек тепловыделяющих элементов эти активные вещества в первую очередь выходят в первичный контур.

Для предотвращения возможного распространения радиоактивных веществ за пределы герметичного первичного контура все оборудование и трубопроводы ЯЭУ ледокола «Ленин», постоянно или периодически заполняемые активной водой, находятся в помещениях так называемого центрального отсека (ЦО), представляющего собой зону строгого режима. Этот отсек изолирован от прочих помещений судна герметичными переборками. Сообщение с ЦО осуществляется только через блокирующий санпропускник с соблюдением соответствующего режима и тщательного радиометрического контроля. ЦО имеет специальную разомкнутую систему вентиляции, изолированную от прочих на судне. Весь воздух из помещений ЦО выбрасывается в грот-мачту с общим расходом $\sim 80\,000$ *м*^{3}/*ч*.

Опыт испытаний и эксплуатации ледокола показал достаточную эффективность такой компоновки радиационно опасного оборудования. Швартовые испытания ЯЭУ ледокола «Ленин» на мощности до 40% производились в черте крупнейшего города Ленинграда. В районе места испытаний была развернута широкая сеть пунктов радиометрического контроля за внешней средой, включая воздух, воду, флору и фауну. Сопоставление результатов измерений до начала испытаний с результатами, полученными во время испытаний и после них, показало, что не наблюдалось каких-либо превышений фоновых значений активностей. Наиболее характерные данные приведены на рис. 3 и 4 для концентрации бета-активных веществ в атмосферном воздухе и в воде Невы [11]. В этих исследованиях было установлено, что колебания уровней удельной активности проб атмосферного воздуха, воды Невы и гидробионтов в районе стоянки атомного ледокола «Ленин»

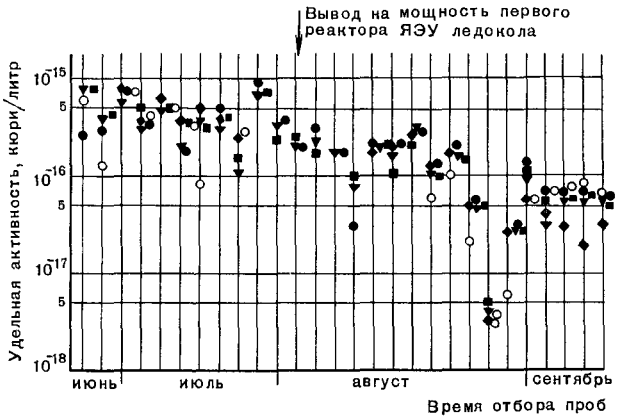


Рис. 3. Концентрация долгоживущих бета-активных аэрозолей в атмосферном воздухе в период швартовых испытаний атомного ледокола «Ленин» (на расстоянии 500—700 *м* от места стоянки)

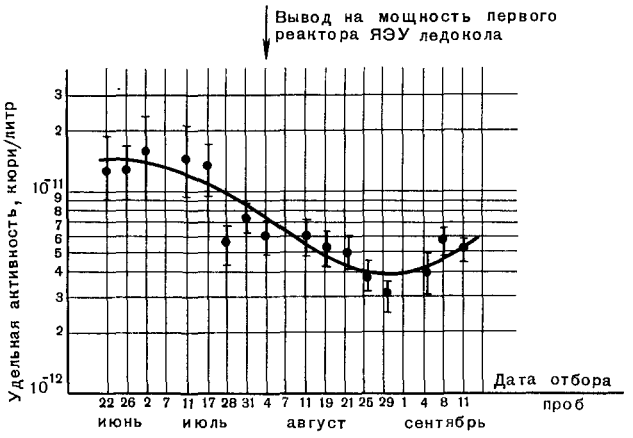


Рис. 4. Концентрация бета-активных веществ в воде Невы за период с 22 июня по 11 сентября 1959 г. (средние величины по всем пунктам)

за период июня — сентября 1959 г. носили сезонный характер, не были связаны с местоположением судна и не зависили от работы ЯЭУ. Характерно, что в соответствии с сезонными колебаниями естественной активности [12] более высокие значения удельной активности проб воздуха, воды и гидробионтов отмечались до начала испытаний ЯЭУ.

активными изотопами инертных газов. Расчеты предельно допустимых концентраций (ПДК) радиоактивных загрязнений для воздуха служебных помещений, основанные на реальных составах радиоактивных ядер в воздухе помещений ЦО, дают значения $\sim 10^{-9}$ *кюри/л*.

При кратковременных течах воды из пер-

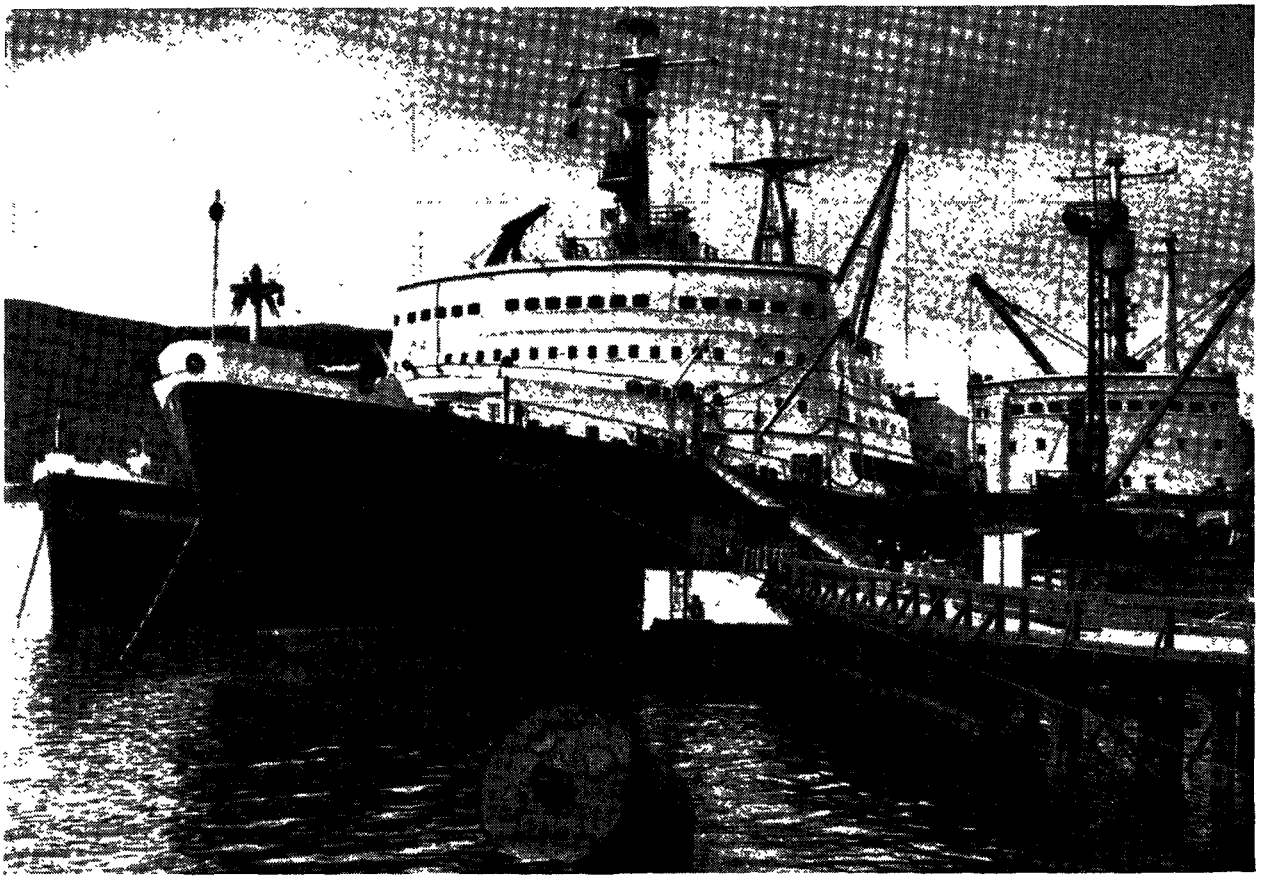


Рис. 5. Атомный ледокол «Ленин» в пункте базирования

Аналогичные выводы получены во время ходовых испытаний ледокола в Балтийском море и в период эксплуатации его на трасе Северного морского пути. В связи с этим было признано возможным базировать атомный ледокол «Ленин» в порту крупного города Мурманска, расположенном в узости (Кольский залив) с интенсивным судоходством.

В период четырехлетней эксплуатации ЯЭУ ледокола имело место несколько случаев утечки воды первичных контуров через разъемные соединения клапанов системы дренажа и подпитки. При этом незначительная часть воды превращалась в пар, который распространялся по помещениям ЦО. Радиометрические анализы паро-воздушной смеси показали, что активность ее более чем на 99,9% определена радиовичного контура интегральная активность аэрозольных долгоживущих продуктов деления, выбрасываемых из грот-мачты ледокола, не превышала 1 *мкюри*. В то же время удельная активность газов на выходе из грот-мачты составляла величину порядка 10^{-8} *кюри/л*. Выброс воздуха из грот-мачты производится со скоростью ~ 20 *м/сек*. Это способствует интенсивной турбулизации и размешиванию активного воздуха в окружающей атмосфере. В связи с этим удельная активность воздуха на расстояниях в несколько десятков метров от верха грот-мачты уменьшается до величин существенно меньше ПДК даже в случаях протечек воды первичного контура.

При нормальном режиме работы ЯЭУ ледокола «Ленин» удельная активность воздуха на

выходе из грот-мачты чрезвычайно мала и лежит ниже предела измерений установленного на судне высокочувствительного прибора ($2 \cdot 10^{-10}$ *кюри/л*).

Жидкие и твердые радиоактивные отходы, образующиеся при работе судовой ЯЭУ ледокола, собираются и хранятся на борту судна в специальных емкостях и хранилищах в зависимости от их удельной активности и физико-химического состава. В пункте базирования отходы передаются на суда специального назначения (рис. 5) для дальнейшей переработки и захоронения в особо отведенных для этого районах.

Накопленный опыт эксплуатации позволяет утверждать, что ЯЭУ ледокола «Ленин» обладает высокоэффективной системой радиационной безопасности, благодаря чему эксплуатация этого судна полностью безопасна для мест стоянки и базирования ледокола и каравана следующих за ним судов и для пищевых и водных ресурсов моря.

ЗАКЛЮЧЕНИЕ

Нами рассмотрены главные вопросы безопасности эксплуатации ЯЭУ ледокола «Ленин», которые обсуждаются в рекомендациях по ядерным судам приложения «С» Международной конвенции 1960 г. по охране человеческой жизни на море [13].

Из анализа данных по аварийности на мировом флоте и опыта эксплуатации ледокола «Ленин» сделан вывод, что ЯЭУ может быть надежным и безопасным судовым двигателем на ледоколах. Этот вывод может быть распространен и на другие атомные суда, если их корпус будет обладать достаточной непотопляемостью и способностью защищать ЯЭУ от значительных повреждений при столкновении с другими судами, что имеет место, например, на грузо-пассажирском судне «Саванна». Заход в порты атомных судов, не удовлетворяющих этим требованиям, следует считать небезопасным.

Установлено, что эксплуатация ЯЭУ ледокола «Ленин» не сопровождается радиационной опасностью для каравана следующих за ним судов, для населения порта, в котором может находиться ледокол, и для пищевых и водных ресурсов моря. В свете рассмотренных пунктов ЯЭУ ледокола «Ленин» соответствует рекомендациям Международной конвенции 1960 г.

В Советском Союзе ведется проектирование новых атомных ледоколов, применение которых в Арктике, как это показал опыт эксплуатации ледокола «Ленин», весьма целесообразно и позволит существенно ускорить развитие Советского Севера.

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ABSTRACT—RÉSUMÉ—АННОТАЦИЯ—RESUMEN

A/378 USSR

Atomic ships safety problems

By V. I. Neganov *et al.*

The paper presents calculations and theoretical considerations, on the basis of which the feasibility of implementing in the icebreaker *Lenin* the basic recommendations relating to atomic-powered ships, embodied in the International Convention on the Safety of Life at Sea, 1960, is analysed. Specific data obtained during practical operation of the icebreaker's atomic power plant, which confirm the soundness of the theoretical calculations, are also given.

The paper discusses the possible radiation hazards that might arise in the event of an atomic-powered ship colliding with other vessels. It presents calculations concerning the possible escape of radioactive fission products from the primary circuit in the case of moderate leaks, and in such situations as that of a burst pipe in the primary circuit, when the reactor core may

be left without water. The extent of damage to the core that may be caused by afterheat release in the uranium is investigated.

A/378 URSS

Problèmes de sécurité relatifs aux navires atomiques

par V. I. Neganov et *al.*

Le rapport présente les calculs et études théoriques qui ont servi de base à l'analyse de la possibilité d'appliquer sur un bateau du type du brise-glace *Lénine* les recommandations fondamentales sur les navires atomiques formulées par la Convention internationale pour la sauvegarde de la vie humaine en mer de 1960. L'expérience pratique du fonctionnement de l'installation atomique du brise-glace a fourni des données concrètes confirmant le bien-fondé des calculs théoriques.

On examine les conséquences du point de vue du danger d'irradiation que pourrait avoir une collision du navire atomique avec d'autres navires. On présente des calculs sur la libération possible de produits de fission actifs du circuit primaire dans le cas de fuites modérées, ainsi que dans le cas, par exemple celui de la rupture d'une conduite du circuit primaire, où le cœur du réacteur peut se trouver privé d'eau. On étudie l'ampleur de la destruction possible du cœur du réacteur du fait du dégagement de chaleur résiduelle dans l'uranium.

A/378 URSS

Problemas de seguridad de los barcos atómicos

por V. I. Neganov et *al.*

En la memoria se presentan los cálculos y las consideraciones teóricas sobre cuya base se analiza un barco del tipo del rompehielos *Lenin* puede cumplir las principales recomendaciones, relativas a los barcos atómicos, expuestas en la Convención international sobre la seguridad de la vida en el mar, de 1960. Deducidos de la práctica de explotación de la instalación atómica del rompehielos *Lenin* se presentan datos concretos que confirman las previsiones teóricas.

Se estudia el posible riesgo de irradiación en caso de choque del barco atómico con otros barcos. Se calculan las posibilidades de escape de los productos de fisión radiactivos del circuito primario, en caso de escapes de poca importancia, y en situaciones análogas a la de rotura de la tubería del circuito primario, como consecuencia de la cual el núcleo puede quedar sin agua. Se estudian las proporciones de la posible destrucción del núcleo debida al calor que quedaría entonces en el uranio.

Record of Session 3.6

Reactor safety; location and containment; reactor tests

Chairman: G. C. Laurence (Canada)

Paper P/611 (presented by S. O. W. Bergström)

DISCUSSION

W. B. COTTRELL (United States of America): To most nuclear safety specialists exclusively concerned with the protection of the public against the potential hazards of radioactivity which might be released during operation of a nuclear facility, it may come as a revelation that the margin of safety between the accepted maximum levels of non-radioactive pollutants and existing levels is often so small. The authors of the paper are to be commended for bringing this information to the attention of the Conference. Certainly, nuclear energy cannot be seen in proper perspective in our society until its hazards are evaluated on a comparable basis with other hazards to which society is subjected.

The question I have on this paper concerns the fact that accidents in nuclear facilities are apparently not considered in reactor siting. Is that correct? Since more activity would be released in an accident than during normal operation what is the justification for not considering accidents in siting?

S. O. W. BERGSTRÖM (Sweden): Nothing but lengthy operational experience can give us information on the most important problem—the probability of really large accidents and their environmental effects. Until such experience has been acquired we have to use our judgement as to how potential hazards should be balanced against those of normal operation. Compared with the way society accepts other large potential hazards and their effects, I am afraid that nuclear energy is unduly hampered by safeguards against big, low-probability accidents. It should be pointed out that the words "normal operation" in our paper include such abnormal situations as running with rather a large number of fuel pin-holes, economically acceptable heavy-water leakage, and so on.

However, even using the concept of maximum credible accidents in the usual way, engineering safeguards should make it possible to place nuclear power stations where they ought to be located for conventional reasons. As is pointed out in our paper, the sites of interest in Sweden up to 1980 will be in areas where safety against large nuclear accidents is no problem.

Paper P/82 (presented by F. de Vathaire)

DISCUSSION

F. C. BOYD (Canada): Who does the hazards analysis work in France and to whom do they report?

F. de VATHAIRE (France): Reactor safety studies are carried out by the various bodies responsible for the projects, i.e., Commissariat à l'Energie Atomique, Electricité de France and industrial concerns. These bodies submit safety reports for analysis by a Reactor Safety Sub-Committee in the CEA at Saclay. The Sub-Committee, presided over by Mr. Bourgeois also checks on the safety of reactors in operation. A group of 15 engineers and physicists, under my direction, deals with this work.

F. C. BOYD (Canada): It is stated that it is possible to protect the filter system on pressurization of the containment. How is this done?

F. de VATHAIRE (France): If the containment stays open at the moment of depressurization of the primary circuit, the pressure rise in the containment can be kept low and the filters can be protected against stress by a suitable aerodynamic system arranged so as to reduce the pressure difference across the filters. Such a system is being considered for EL4.

W. K. DAVIS (United States of America): You say you consider an accident which would release a large amount of radioactivity to be essentially incredible. I am wondering whether this means that gas-cooled graphite reactors can be sited in any location or whether other safety considerations are involved in siting.

F. de VATHAIRE (France): French reactors of the graphite/gas type are not equipped with a leak-proof containment. Only the reactor EDF1 has been fitted with an external sphere, but the latter does not act as a leak-proof containment, because safety studies have shown this to be useless. Studies have shown that in accidents involving the worst possible pipe breakages the temperatures of cans and graphite do not exceed safe limits, so that any important emission of fission products is extremely unlikely; this favourable circumstance is due to the low specific power in the fuel and the large calorific mass of the graphite moderator block.

Graphite/gas reactors could thus, theoretically, be sited near towns, but it would be wise, for the time being, to wait until experience has been gained on the large power stations now being constructed at distances of several tens of kilometres from large towns.

H. KOUTS (United States of America): It is stated in the paper that future large fast reactors can be contained by simple means, particularly as regards the mechanical effects of credible power excursions. Could anybody elaborate on this?

G. DREVON (France): Taking the most pessimistic view with regard to radioactivity coefficients, the most serious accident which seems possible on the basis of preliminary studies of industrial power stations is that of relatively slow melting of part of the fuel. In this connexion I would refer you to report CEA-2554. If, under certain conditions, the sodium in the central part of the core vaporized there could be considerable introduction of radioactivity. From studies carried out so far, however, it seems that the rate of introduction of this radioactivity could not be high enough to cause an explosive accident.

Paper P/182

DISCUSSION

J. TADMOR (Israel): Your paper mentions that the performance of filters used in contained and vented reactors has been improved. Recent experiments performed on the release of fission products from reactor fuel revealed that some of the fission-product particles are in the Angström size range rather than in the micron range, so I would like to ask whether anything has been done to improve the efficiency of absolute filters for particles in the Angström size range. To my best knowledge the absolute filters available at present are efficient only for particle sizes in the micron range.

F. R. FARMER (United Kingdom): I believe that, in the experiments you refer to, UO_2 was heated to very high temperatures—higher than 2 000 °C. An accident analysis for the AGR gives confidence that this temperature will not be reached. We have heated irradiated UO_2 fuel in the range 1 600–2 000 °C and released 4 000 curies in an experimental facility. This has shown that ^{131}I can be retained to an efficiency in the range 10^3 to 10^4.

B. G. DUBOVSKY (USSR): Could the authors of papers P/182 and P/275 give their views on the different requirements for siting power reactors in the United Kingdom and the United States?

F. R. FARMER (United Kingdom): There is no really fundamental difference between the safety approach of the UK and that of the USA. There are differences in design solutions owing to the choice of different reactors. If the UK had chosen the water system, the safety features in the early development would have been very close to those adopted by the USA. During the course of development some differences are bound to emerge and these are a healthy sign.

C. K. BECK (United States of America): I believe the conclusions reached are not very different in the two cases, though the reactors have generally been quite different. In the United States we have developed extremely valuable guides based on the maximum credible accident. Consistent and uniform guidance has been given to the hundreds of different organizations interested in nuclear reactors and reactor designers have had an incentive to improve safety and safeguards systems, inasmuch as the safety distances could be reduced. The UK has employed an entirely different procedural approach.

Paper P/380 (presented by V. A. Sidorenko)

DISCUSSION

J. LIEBERMAN (United States of America): The design and operation of the reactivity controls described in your paper are interesting, and the criteria involved seem quite similar to those common in other countries. However, I would like to ask what consideration is given to the possibility of other types of reactor accident, such as loss of coolant. Also, what provisions are made for protecting the public in the event of accidents involving fission product release?

V. A. SIDORENKO (USSR): My paper deals with safety regulations for preventing uncontrollable nuclear excursions and unfortunately I cannot give a general survey of other types of dangerous situation. However, I might perhaps mention the protective measures applied at the Novovoronezh power station. All the primary circuit equipment is enclosed in reinforced concrete boxes lined with steel. In case of leakage in the primary circuit, all the active products remain within this space. The containers also include a filter system in the outlet pipe, which ensures that permissible concentrations of active products in the surrounding air are not exceeded.

B. G. DUBOVSKY (USSR): I should like to add that at the I. V. Kurchatov nuclear power station at Beloyarsk, USSR, the piping of the primary circuit of the high specific-power reactor and part of the piping of the secondary circuit (which could become radioactive as the result of an incident) are also placed in closed, ventilated boxes with thick concrete or cast-iron walls for biological shielding. The central reactor hall is also ventilated; the air directly above the upper part of the reactor is controlled. There is a central

dosimetric panel from which it is easy to locate an area of increased activity and to determine the air contamination in all associated compartments. Detailed plans have been made for the evacuation of personnel in the event of different types of incident. Similar measures have been planned for all the reactors in operation. The reactors are located at a reasonable distance from settled areas.

J. STORRER (Belgium): I should like to know whether the regulations mentioned in your paper permit fuelling of water reactors without "safety elements", e.g., when sufficient subcriticality is ensured by soluble poison, such as boric acid, and when the measuring instruments are continuously observed from the start of loading operations.

V. A. SIDORENKO (USSR): The regulations require that at least two "safety elements" must be in the ready-to-operate position before the reactor goes into the critical state, regardless of whether there is additional provision for compensating reactivity, in the form of boric acid solution, or whether the reactor is controlled in the entire subcritical range.

R. F. JACKSON (United Kingdom): Paragraph 4 of the safety regulations set out in your paper appear to require that during a melt-down accident there shall be a safety factor of 2 between the largest mass of molten fuel which can assemble at any point and a critical mass. Are you confident that this requirement will be met in the design of the proposed Soviet fast reactors of 350 and 1 000 MW(e) and that any possibility of nuclear energy release in a melt-down accident can be discounted, both during the melting process and in the final catchpot arrangement below the core?

V. A. SIDORENKO (USSR): These safety regulations were approved by the State Committee for Atomic Energy and I have heard of no objections on the part of the fast reactor designers. I do not know how the designers will overcome this difficulty but I am confident they will manage.

F. DUHAMEL (France): I should like to ask the authors or any of their colleagues whether technical restrictions are imposed in the USSR governing reactor installation near urban populations. For example, is any minimum distance from reactor to populated area prescribed? I ask this because the Soviet SBR5 reactor is situated in the vicinity of a built-up area.

B. G. DUBOVSKY (USSR): The SBR5 fast reactor (5 000 kW) is in fact located within the limits of the town of Obninsk, but it is a research reactor. In the USSR, fast and thermal reactors for nuclear power stations are already being built near built-up areas, and Soviet scientists and engineers are working to increase the safety of bigger power reactors, so that they can be located near actual towns.

Paper P/28 (presented by F. C. Boyd)

DISCUSSION

A. BULL (Norway): Do you anticipate any safety difficulties in carrying out maintenance work on the containment?

F. C. BOYD (Canada): In the case of the Douglas Point nuclear station, for example, all entrances to the containment building have air-locks, and these are large enough for all normal maintenance, so that no breach of the containment occurs as a result of maintenance. If much larger equipment had to be moved in or out this would be considered an exceptional situation and special procedures would be prepared.

K. MAHMOUD (United Arab Republic): In your paper, you state that "... the exclusion area is intended to protect the public in any accident ...". Is that the only precaution in respect of an accident, or are special emergency plans to be studied for the different types of Canadian reactor? If so, what are the important criteria to be considered?

F. C. BOYD (Canada): Taking the NPD nuclear station as an example, emergency procedures have been prepared for the protection of the plant personnel and the surrounding population. Plant-evacuation drill is conducted periodically. Regular contact is maintained with police, medical officers of health, etc. to prepare for evacuation of the public, if necessary. Monitors are installed at different critical points outside the exclusion area. Monitoring teams are also available at short notice.

Paper P/275

DISCUSSION *

F. C. BOYD (Canada): Your paper states that 3 power reactors were located closer to population centres than otherwise would be allowed because "certain specific safeguard systems were incorporated into the design". Could you give details of these specific systems?

C. K. BECK (United States of America): The Malibu site in the city of Los Angeles has been approved for a large power reactor because of certain safeguards in the design: double containment system, improved emergency cooling systems, and other safety features. These, we believe, would so reduce the consequences of any serious accident that some reduction in distance requirements can be justified.

J. WENGLER (Federal Republic of Germany): What engineered safeguards do you propose for steam-tube

* See also discussion on paper P/182.

rupture between the containment and the turbine in the case of a direct-cycle boiling-water reactor?

C. K. BECK (United States of America) Appropriate isolation valves fitted in the primary lines at the containment wall will close in the event of rupture of the outside lines.

V. A. SIDORENKO (USSR): Does experience with operation of American reactors support the idea that the worst type of accident is that caused by loss of coolant? There may be other types of accident which are not so bad from the point of view of actual effects, but which are statistically more likely to occur, for example the uncontrolled withdrawal of a rod.

C. K. BECK (United States of America): Safety research has shown that in most water reactors the worst accidents against which safety systems of the consequence-limiting type have to be provided are those caused by loss of coolant. These accidents are also used to judge the adequacy of sites. If there is adequate protection against these maximum credible accidents with such reactors, there will also be adequate protection of the public against lesser, perhaps more probable, accidents.

However, in some reactors, for instance fast reactors and at an earlier date water reactors, reactivity accidents can be regarded as the maximum credible ones.

Papers P/714 (presented by D. A. Davies) and P/793 (presented by V. Surya-Rao)

There was no discussion of these papers.

Paper P/378 (presented by V. G. Pologikh)

DISCUSSION

A. BULL (Norway): In the event of a serious reactor accident on a nuclear ship, involving the possible release of radioactive fission products, the removal of the ship from the harbour or from nearby populated areas is important. However, in the case of a nuclear-propelled icebreaker, this operation can be delayed or made impossible by ice conditions. I would like to ask your views on this problem.

B. G. POLOGIKH (USSR): The *Lenin* icebreaker is based on Kola Bay, where there is no ice.

W. VINKE (Netherlands): Your paper states that operation of the *Lenin* icebreaker does not present a radiation hazard for harbour populations. One of the protective measures that can be adopted when the ship is visiting ports or passing through channels is to reduce power for a certain period before entering. The length of this period is a matter for discussion and depends on how safe one wants to be. One could stipulate say 6, 24 or 48 hours or even that the reactor must be stopped and depressurized.

Could you give an indication of the philosophy on this matter in the USSR, and what criteria have been adopted to guarantee the future of nuclear ship propulsion?

B. G. POLOGIKH (USSR): When a nuclear vessel approaches a harbour, the power of the plant is always reduced. The necessity of reducing pressure in the primary circuit depends on the equipment: in certain types of water-cooled reactor, some lowering of pressure will probably be required, but of course pressure need not be completely reduced.

U. M. STAEBLER (United States of America): The statistical data on ship accident rates used in paper P/378 include craft as small as 100 gross registered tons and do not adequately differentiate between minor damage and total loss. According to our sources in the Maritime Administration, the loss rate in 1962 for ships over over 500 gross registered tons was about 0.5% and for ships over 1 000 gross registered tons it was about 0.26%. The average size of new American merchant ships is around 9 000 gross registered tons and any nuclear ships would be of this size or larger. Thus, I suggest that the parameters chosen should be significantly modified in this study before making any judgements about potential hazards of using nuclear power in merchant ships.

With respect to the NS *Savannah*, I wish to call attention to just three of the special safety features in her design:

(*a*) A reactor containment vessel separate from the hull structure;

(*b*) Additional structure outboard of the containment to minimize the effects of collision;

(*c*) A foundation designed to minimize the effects of grounding and to restrain the containment vessel and its contents with the ship in extreme positions.

B. G. POLOGIKH (USSR): For the time being, I cannot say anything definite about future nuclear plants on icebreakers, but there can be no doubt that improvements will be introduced in the light of operational experience with the *Lenin* icebreaker.

GENERAL DISCUSSION

A. K. GANGULY (India): I should like to comment as follows. The siting criteria and actual siting practices adopted in different countries seem rather arbitrary. This arbitrariness is due to differing national requirements, but I think it is nevertheless possible to ensure some standardization in siting and safety procedures: sites and reactor systems are taken as independent parameters in this drill.

A site can be defined or graded according to acceptable release in an emergency, or acceptable radiological site burden. This can be done on the basis of site

studies and national safety codes. Reactor systems can then be graded according to their containment capabilities. The third step is then to consider the combination of given sites and given reactor systems. One can now apply judgement in adjusting or modifying the two to conform to the national safety code.

L. CAVE (United Kingdom): With regard to the use of the maximum credible accident concept, I should like to comment from the point of view of the designer. At any given time in the development of a particular reactor system, when a new order is about to be placed, there must be agreement between the designer, the purchaser and the licensing authority as to the nature of the faults against which the designer must provide protection and the effectiveness of that protection. Unless this is done, the detailed design cannot proceed in a sensible fashion. In the area of procurement and design there must therefore be a clear definition of the "maximum accident to be considered in design": this concept is obviously related to that of the maximum credible accident and probably this explains why the latter concept has been so widely used.

As experience with the system is gained the "maximum accident to be considered in design" may well change, with resulting modifications of the emergency equipment or changes in operating conditions of existing reactors of the same type.

V. NADEINSKI (IMCO): I should like to make a statement regarding the work of the Intergovernmental Maritime Consultative Organization. IMCO is a specialized agency of the United Nations concerned with international maritime affairs, its objective being to facilitate co-operation between Governments in shipping matters. One of its main aims is to encourage general adoption of the highest practicable standards of maritime safety and efficient navigation.

IMCO has a special interest in the problems of the peaceful uses of atomic energy and, in particular, in the fields of nuclear propulsion and the carriage of radioactive substances. It administers the International Convention for the Safety of Life at Sea, 1960, which will come into force on 25 May 1965, and, in this connexion, has a special responsibility with regard to nuclear ships. The 1960 Safety Convention applies to all nuclear ships, except warships. Chapter VIII of the Convention is devoted to nuclear ships and contains a number of regulations which are mostly limited to questions of principle and procedure. In accordance with these regulations, the design, construction and standards of inspection, as well as the assembly of the reactor installation of a nuclear ship, are to be approved by the governmental authority; nuclear ships should be surveyed, not less than once a year, and should carry a special Nuclear Ship Safety Certificate which, among other things, will state that the ship, being a nuclear one, complies with all requirements of Chapter VIII of the Convention and conforms to the Safety Assessment approved for the ship. All other chapters of the 1960 Convention also apply to nuclear ships, except as modified in Chapter VIII.

The International Conference on Safety of Life at Sea, 1960, which prepared the Safety Convention, adopted a number of recommendations applicable to nuclear ships, in order to provide guidance for Governments in the application of the regulations included in the Safety Convention, and to draw attention to the main problems which exist at the present stage of technical development. These recommendations cover various subjects, such as the general safety of nuclear ships, general requirements for nuclear-powered plant, protection and enclosure of reactor installations, shielding and radiation safety, radioactive wastes, fuelling and maintenance, manning, operating manuals, safety assessment and special control. These subjects will have to be dealt with by IMCO in co-operation with the IAEA.

Another aspect of the peaceful uses of atomic energy with which IMCO is concerned is the carriage of radioactive substances by sea. The increasing use of radioactive sources for industrial and medical purposes and the construction of reactors in the developing countries both lead to increased shipments of radioactive material. In accordance with the relevant Recommendation of the 1960 Safety Conference, one of IMCO's working groups is preparing a code of safe practice for the transport by sea of all dangerous goods, including radioactive substances. The code will be based eventually on the recently revised transport regulations of the IAEA, and also on existing maritime regulations. This international code will contribute to the extension of the peaceful uses of atomic energy.

There are only two merchant nuclear ships in operation at the moment—the icebreaker *Lenin* and the passenger ship *Savannah*—but some other nuclear ships are under construction or design. It may well be that, in a comparatively short time, the nuclear ship will become conventional. As the number of nuclear ships increases, so will the need arise for the development of more detailed international regulations for nuclear ships. This is one of the tasks which IMCO will have before it.

Compte rendu de la séance 3.6

Sûreté des réacteurs; implantation et confinement; essais d'épreuve

Président: G. C. Laurence (Canada)

Mémoire P/611 (présenté par S. O. W. Bergström)

DISCUSSION

W. B. COTTRELL (Etats-Unis d'Amérique): La plupart des spécialistes de la sûreté nucléaire, qui s'occupent exclusivement de protéger le public contre les risques que pourrait présenter la radioactivité libérée au cours du fonctionnement d'une installation nucléaire, seront sans doute surpris de constater que la marge de sécurité entre les niveaux maximaux admissibles pour des agents de pollution non radioactifs et les niveaux existants soit souvent si faible. Il faut féliciter les auteurs de ce mémoire d'avoir attiré l'attention de la Conférence sur cette question. Il est certain que l'énergie nucléaire ne peut être jugée correctement dans notre société tant que les risques qu'elle présente ne seront pas évalués avec les mêmes critères que les autres risques auquels la société est exposée.

La question que je voudrais poser à propos de ce mémoire concerne le fait que les accidents pouvant survenir dans des installations nucléaires ne sont apparemment pas envisagés au moment du choix du site. Ceci est-il exact? Comme une quantité plus importante de radioactivité serait libérée au cours d'un accident qu'en fonctionnement normal, quelle justification donne-t-on pour ne pas envisager les accidents en choisissant le site?

S. O. W. BERGSTRÖM (Suède): Il n'y a qu'une longue expérience de fonctionnement qui puisse nous fournir des renseignements sur le problème le plus important, la probabilité des accidents réellement graves et leurs effets aux environs du site. Tant que nous ne disposons pas de cette expérience, nous devons utiliser notre jugement pour évaluer les risques potentiels par rapport à ceux du fonctionnement normal. Par comparaison avec la façon dont la société accepte d'autres risques potentiels importants et leurs conséquences, je crois que l'énergie nucléaire est anormalement gênée par les précautions prises contre de graves accidents à faible probabilité. Il faut souligner que le terme « fonctionnement normal » dans notre mémoire comprend des situations anormales telles que le fonctionnement d'un réacteur avec un assez grand nombre de petits trous dans les éléments de combustible, des fuites d'eau lourde économiquement acceptables, etc.

Cependant, même en utilisant le concept de l'accident maximal prévisible dans son sens habituel, les précautions techniques devraient permettre d'installer des centrales nucléaires aux endroits où elles devraient être situées pour des raisons classiques. Comme notre mémoire l'indique, les sites intéressants en Suède jusqu'en 1980 seront dans des régions où la protection contre de graves accidents nucléaires ne pose pas de problèmes.

Mémoire P/82 (présenté par F. de Vathaire)

DISCUSSION

F. C. BOYD (Canada): Qui s'occupe de l'analyse de la sûreté des réacteurs en France et à qui sont adressés les rapports?

F. DE VATHAIRE (France): Les études de sûreté des réacteurs sont faites par les divers organismes responsables des projets, c'est-à-dire le Commissariat à l'énergie atomique, l'Electricité de France et les sociétés industrielles. Ces organismes soumettent les rapports de sûreté à la Sous-Commission de sûreté des piles, à Saclay. La Sous-Commission, présidée par M. Bourgeois, vérifie également la sûreté des réacteurs en fonctionnement. Ce travail est accompli, sous ma direction, par un groupe de 15 ingénieurs et physiciens.

F. C. BOYD (Canada): Il est dit dans ce mémoire qu'il est possible de protéger le système de filtres lorsqu'il y a pressurisation de l'enceinte. Comment est-ce réalisé?

F. DE VATHAIRE (France): Si l'enceinte reste ouverte au moment de la dépressurisation du circuit primaire, ou peut limiter l'augmentation de pression dans l'enceinte et protéger les filtres contre les efforts à l'aide d'un circuit aérodynamique adéquat comportant une section de passage des filtres assez grande. Des venturis de protection des filtres ne sont pas alors nécessaires. Cette solution n'a pas été adoptée pour EL4.

W. K. DAVIS (Etats-Unis d'Amérique): Vous dites que vous jugez tout à fait incroyable un accident qui libérerait une quantité importante de radioactivité. Je me demande si cela veut dire que les réacteurs à gaz et graphite peuvent être situés n'importe où, ou si d'autres considérations interviennent dans le choix du site.

F. DE VATHAIRE (France): Les réacteurs français du type gaz-graphite n'ont pas d'enceinte étanche. Seul le réacteur EDF1 a été équipé d'une sphère externe, mais cette dernière ne joue pas le rôle d'une enceinte étanche, parce que les études de sûreté ont montré que c'était inutile. Les études ont montré que dans les accidents avec les pires ruptures de canalisations les températures des gaines et du graphite ne dépassent pas les limites de sûreté, ce qui rend extrêmement improbable l'émission de produits de fission; ces circonstances favorables sont dues à la faible puissance spécifique du combustible et à la grande masse calorifique du modérateur de graphite. On pourrait donc théoriquement mettre les réacteurs gaz-graphite au voisinage des villes, mais il est prudent, pour le moment, d'attendre l'expérience que nous donnerons les grandes centrales en cours de construction à des distances de quelques dizaines de kilomètres de villes importantes.

H. KOUTS (Etats-Unis d'Amérique): Il est indiqué dans le mémoire que les grands réacteurs rapides de l'avenir pourront être protégés par des moyens simples, en particulier en ce qui concerne les effets mécaniques des sautes de puissance que l'on peut envisager. Quelqu'un pourrait-il commenter cette déclaration?

G. DREVON (France): En évaluant les coefficients de réactivité de la façon la plus pessimiste, l'accident le plus sérieux qui semble possible sur la base des études préliminaires des centrales industrielles est la fusion relativement lente d'une partie du combustible. A ce sujet, je vous prie de vous reporter au rapport CEA 2554. Si, dans certaines conditions, le sodium de la partie centrale du cœur était vaporisé, il pourrait y avoir une introduction considérable de réactivité. Les études faites à ce jour semblent cependant indiquer que la vitesse d'introduction de cette réactivité ne serait pas suffisante pour provoquer un accident explosif.

Mémoire P/182

DISCUSSION

J. TADMOR (Israël): Votre mémoire mentionne que les performances des filtres utilisés dans les réacteurs à enceinte étanche et dans les réacteurs avec enceinte non étanche ont été améliorées. Des expériences récentes sur la libération des produits de fission à partir de combustibles de réacteur ont révélé que certaines des particules de produits de fission ont des dimensions de l'ordre de l'angström plutôt que de l'ordre du micron. Je voudrais donc demander si on a cherché à améliorer l'efficacité des filtres absolus pour les particules dans le domaine de l'angström. A ma connaissance, les filtres absolus dont on dispose actuellement ne sont efficaces que pour des tailles de particules dans le domaine du micron.

F. R. FARMER (Royaume-Uni): Je crois que dans les expériences auxquelles vous pensez, du UO_2 était porté à de très hautes températures, supérieures à 2 000 °C. L'analyse de sûreté pour l'AGR garantit que cette température ne sera pas atteinte. Nous avons chauffé du combustible en UO_2 irradié dans l'intervalle 1 600–2 000 °C et libéré 4 000 Ci dans une installation expérimentale. Cette expérience a montré que l'iode 131 peut être retenu avec une efficacité de 10^3 à 10^4.

B. G. DUBOVSKY (URSS): Les auteurs des mémoires P/182 et P/275 pourraient-ils donner leur point de vue sur les différentes exigences relatives aux sites pour les réacteurs de puissance au Royaume-Uni et aux Etats-Unis?

F. R. FARMER (Royaume-Uni): Il n'y a pas de différence fondamentale entre les conceptions de la sûreté des réacteurs au Royaume-Uni et aux Etats-Unis. Il y a des différences dans les solutions techniques en raison du choix de différents types de réacteurs. Si le Royaume-Uni avait choisi le système à eau, les aspects de sûreté au début du développement auraient été très proches des solutions adoptées aux Etats-Unis. Il faut s'attendre à l'apparition de quelques différences en cours de développement, ce qui est bon signe.

C. K. BECK (Etats-Unis d'Amérique): Je crois que les conclusions ne sont pas très différentes dans les deux cas, bien que les réacteurs soient généralement très différents. Aux Etats-Unis nous avons élaboré des guides très utiles, basés sur l'accident maximal prévisible. Des directives logiques et uniformes ont été données aux centaines d'organisations différentes qui s'intéressent aux réacteurs nucléaires, et ceux qui conçoivent les réacteurs ont été poussés à accroître la sûreté et les systèmes de protection, dans la mesure où les distances de sûreté pouvaient être réduites. Le Royaume-Uni a abordé ce problème de façon entièrement différente.

Mémoire P/380 (présenté par V. A. Sidorenko)

DISCUSSION

J. LIEBERMAN (Etats-Unis d'Amérique): La conception et le fonctionnement des contrôles de réactivité décrits dans votre mémoire sont intéressants, et les critères correspondants semblent très voisins de ceux qui sont usuels dans d'autres pays. Je voudrais cependant vous demander si vous envisagez la possibilité d'autres types d'accident de réacteur, comme la perte de réfrigérant. Et aussi, comment vous envisagez la protection du public en cas d'accidents entraînant la libération de produits de fission.

V. A. SIDORENKO (URSS): Mon mémoire concerne les règles de sûreté destinées à éviter des excursions nucléaires incontrôlables, et je ne peux malheureuse-

ment pas passer en revue les autres types de situations dangereuses. Je peux cependant mentionner peut-être les mesures de protection prises à la centrale de Novovoronej. Toute l'installation du circuit primaire est à l'intérieur de caissons de béton renforcé revêtus d'acier. En cas de fuite dans le circuit primaire, tous les produits actifs restent dans cet espace. Il y a aussi un système de filtres sur la canalisation de sortie, pour que les concentrations admissibles de produits actifs dans l'atmosphère ne soient pas dépassées.

B. G. DUBOVSKY (URSS): Je voudrais ajouter qu'à la centrale nucléaire I.V. Kurchatov de Beloyarsk (URSS) les canalisations du circuit primaire du réacteur à grande puissance spécifique et une partie du circuit secondaire (qui pourrait devenir radioactif à la suite d'un incident) sont aussi placées dans des compartiments fermées, ventilées, avec des parois en béton épais ou en fonte pour assurer la protection biologique. Le hall central du réacteur est aussi ventilé; l'air qui se trouve directement au-dessus de la partie supérieure du réacteur est contrôlé. Il y a un pupitre dosimétrique central d'où l'on peut facilement localiser une zone d'augmentation de l'activité et déterminer la contamination atmosphérique dans tous les compartiments associés. Des plans détaillés ont été faits pour l'évacuation du personnel pour différents types d'incidents. Des mesures semblables ont été prévues pour tous les réacteurs en fonctionnement. Les réacteurs sont situés à des distances raisonnables des régions peuplées.

J. STORRER (Belgique): Je voudrais savoir si les règlements mentionnés dans votre mémoire autorisent le chargement du combustible dans des réacteurs à eau sans « éléments de sûreté », c'est-à-dire avec une sous-criticité suffisante assurée par un poison soluble, comme l'acide borique, et en observant de façon continue les instruments de mesure depuis le début des opérations de chargement.

V. A. SIDORENKO (URSS): Les règlements exigent qu'il y ait au moins deux « éléments de sûreté » en position prêt-à-fonctionner avant que le réacteur n'atteigne l'état critique, indépendamment de précautions additionnelles pour compenser la réactivité, sous forme de solution d'acide borique, et du contrôle du réacteur dans tout le domaine sous-critique.

R. F. JACKSON (Royaume-Uni): Le paragraphe 4 des règles de sécurité mentionnées dans votre mémoire semble exiger que pendant un accident de fusion la plus grande masse de combustible fondu qui peut se rassembler en un point ne soit pas supérieure à la moitié de la masse critique. Espérez-vous pouvoir suivre ce règlement dans l'étude des réacteurs rapides soviétiques de 350 et 1 000 MW(e) et pouvoir assurer qu'il n'y aura pas de dégagement d'énergie nucléaire dans un accident de fusion, soit pendant le processus de fusion, soit dans l'appareil de récupération placé sous le cœur?

V. A. SIDORENKO (URSS): Ces règlements de sécurité ont été approuvés par le Comité d'Etat pour l'énergie atomique et je n'ai pas entendu parler d'objections de la part de ceux qui étudient les réacteurs rapides. Je ne sais pas comment les ingénieurs surmonteront cette difficulté, mais je suis sûr qu'ils y parviendront.

F. DUHAMEL (France): Je voudrais demander aux auteurs ou à leurs collègues si des règles techniques conditionnent en URSS l'installation de réacteurs au voisinage de centres urbains. Par exemple, y a-t-il une distance minimale fixée entre un réacteur et une agglomération? Je pose cette question parce que le réacteur SBR-5 est situé au voisinage d'une agglomération.

B. G. DUBOVSKY (URSS): Le réacteur rapide SBR-5 (5 000 kW) est en fait situé dans la ville d'Obninsk, mais c'est un réacteur de recherche. En URSS, nous construisons déjà des réacteurs rapides et thermiques pour des centrales nucléaires au voisinage d'agglomérations et les savants et les ingénieurs soviétiques travaillent à accroître la sûreté des réacteurs de puissance de plus grande taille, pour permettre de les situer au voisinage des villes.

Mémoire P/28 (présenté par F. C. Boyd)

DISCUSSION

A. BULL (Norvège): Prévoyez-vous des difficultés sur le plan de la sûreté pour les travaux d'entretien sur l'enceinte?

F. C. BOYD (Canada): Dans le cas de la centrale de Douglas Point, toutes les entrées dans l'enceinte de confinement ont des sas, qui sont assez grands pour tout l'entretien normal, de sorte qu'il n'y a pas rupture de l'étanchéité à l'occasion de l'entretien. S'il fallait faire entrer ou sortir de l'équipement beaucoup plus grand, ce serait considéré comme une situation exceptionnelle et on préparerait des procédures spéciales.

K. MAHMOUD (République arabe unie): Dans votre mémoire, vous déclarez que la zone d'exclusion a pour but de protéger le public dans tout accident... S'agit-il de la seule précaution en cas d'accident, ou y a-t-il des plans spéciaux de secours étudiés pour les différents types de réacteurs canadiens? Si la réponse est oui, quels sont les critères importants qu'il faut envisager?

F. C. BOYD (Canada): Pour la centrale NPD par exemple, des procédures de secours ont été préparées pour la protection du personnel de la centrale et des populations voisines. Des exercices d'évacuation de la centrale ont lieu périodiquement. On maintient régulièrement le contact avec la police, les médecins du service de santé, etc., pour préparer l'évacuation du public en cas de besoin. Des détecteurs sont installés en différents points critiques en dehors de la zone d'exclusion. Des équipes de mesure sont prêtes à se déplacer avec un préavis très court.

Mémoire P/275

DISCUSSION *

F. C. BOYD (Canada): Votre mémoire indique que trois réacteurs de puissance ont été installés plus près de centres de population qu'on ne l'aurait normalement autorisé, parce que certains dispositifs spéciaux de protection ont été incorporés dans les plans. Pourriez-vous donner des détails sur ces systèmes spéciaux?

C. K. BECK (Etats-Unis d'Amérique): Le site de Malibu dans la ville de Los Angeles a été approuvé pour un grand réacteur de puissance en raison de certaines garanties dans les plans: double système de confinement, systèmes améliorés de refroidissement de secours et autres facteurs de sécurité. Nous pensons que cet ensemble de mesures réduirait à un tel point les conséquences d'un accident sérieux qu'une certaine réduction des distances requises est justifiée.

J. WENGLER (République fédérale d'Allemagne): Quelles mesures techniques prévoyez-vous en cas de rupture des conduits de vapeur entre l'enceinte et la turbine dans le cas d'un réacteur à eau bouillante à cycle direct?

C. K. BECK (Etats-Unis d'Amérique): Des vannes d'isolement appropriées situées dans le circuit primaire à la paroi de l'enceinte, se fermeront en cas de rupture des conduites extérieures.

V. A. SIDORENKO (URSS): L'expérience de fonctionnement des réacteurs des Etats-Unis confirme-t-elle l'idée que le type d'accident le plus grave est celui qui est provoqué par la perte de réfrigérant? Il peut y avoir d'autres types d'accident, moins graves au point de vue des conséquences, mais plus probables statistiquement, comme par exemple le retrait non contrôlé d'une barre.

C. K. BECK (Etats-Unis d'Amérique): Les études de sécurité ont montré que pour la plupart des réacteurs à eau les accidents les plus graves, contre lesquels il faut disposer de systèmes de sécurité permettant de limiter les conséquences, sont les accidents par perte de réfrigérant. Ces accidents sont ceux que l'on considère également pour décider si un site est adéquat. S'il y a une protection suffisante contre ces accidents « maximaux prévisibles » avec de tels réacteurs, il y aura aussi une protection suffisante du public contre des accidents moins graves, même s'ils sont plus probables.

Cependant pour certains réacteurs, par exemple les réacteurs rapides et, plus immédiatement, les réacteurs à eau ,les accidents de réactivité peuvent être considérés comme les accidents maximaux prévisibles.

* Voir aussi la discussion du mémoire P/182.

Mémoires P/714 (présenté par D. A. Davies) et P/793 (présenté par V. Surya-Rao)

Ces mémoires n'ont pas fait l'objet d'une discussion.

Mémoire P/378 (présenté par B. G. Pologikh)

DISCUSSION

A. BULL (Norvège): En cas d'accident sérieux de réacteur sur un navire nucléaire, avec libération possible de produits de fission radioactifs, il est important d'éloigner le navire du port ou des régions peuplées voisines. Cependant, dans le cas d'un brise-glace à propulsion nucléaire, cette opération peut être retardée ou rendue impossible par les glaces. Je serais heureux d'avoir votre avis sur ce problème.

B. G. POLOGIKH (URSS): Le brise-glace *Lénine* a sa base dans la baie de Kola, où il n'y a pas de glace.

W. VINKE (Pays-Bas): Votre mémoire indique que le fonctionnement du *Lénine* ne présente pas de dangers dus aux rayonnements pour les populations des ports. L'une des mesures de protection que l'on peut adopter quand le navire est en visite dans des ports ou passe dans des chenaux consiste à réduire la puissance pendant un certain temps avant d'y pénétrer. La longueur de cette période est matière à discussion et dépend du degré de sécurité désiré. On peut stipuler 6, 24 ou 48 h, ou même demander que le réacteur soit arrêté et dépressurisé.

Pourriez-vous donner une indication des points de vue adoptés dans ce domaine en URSS, et des critères qui ont été admis pour garantir l'avenir de la propulsion nucléaire des navires.

B. G. POLOGIKH (URSS): Quand un navire nucléaire s'approche d'un port, la puissance du réacteur est toujours réduite. La nécessité de réduire la pression dans le circuit primaire dépend de l'équipement; dans certains types de réacteurs refroidis à l'eau, une certaine diminution de pression sera probablement exigée, mais il est bien entendu qu'il n'est pas nécessaire de réduire complètement la pression.

U. M. STAEBLER (Etats-Unis d'Amérique): Les données statistiques sur les taux d'accidents de navire utilisées dans le mémoire P/378 comprennent des embarcations à partir de 100 tonneaux de jauge pour les plus petites, et ne font pas suffisamment de différence entre les dégâts légers et la perte totale. D'après les chiffres de l'Administration maritime, les taux de perte en 1962 pour les bâtiments de plus de 500 tonneaux de jauge était d'environ 0,5%, et pour les bâtiments de plus de 1 000 tonneaux de jauge d'environ 0,26%. La taille moyenne des nouveaux navires marchands des Etats-Unis est voisine de 9 000 tonneaux de jauge, et les navires nucléaires seraient de cette taille ou plus grands. Je pense donc que les paramètres

choisis devraient être modifiés de façon importante dans cette étude avant de juger des risques d'utilisation de la propulsion nucléaire pour les navires marchands.

En ce qui concerne le NS *Savannah*, je voudrais attirer l'attention sur trois des caractéristiques spéciales de sûreté dans sa conception:

a) Une enceinte de confinement du réacteur distincte des structures de la coque;

b) Une structure additionnelle à l'extérieur de l'enceinte pour limiter les effets d'une collision;

c) Des bases conçues pour minimiser les effets d'un échouage et pour retenir l'enceinte de confinement et son contenu lorsque le navire prend des positions extrêmes.

B. G. POLOGIKH (URSS): Pour le moment, je ne peux rien dire de précis sur les réacteurs futurs des brise-glace, mais il est certain que des améliorations seront apportées à la lumière de l'expérience de fonctionnement obtenue avec le brise-glace *Lénine*.

DISCUSSION GÉNÉRALE

A. K. GANGULY (Inde): Je voudrais faire les observations suivantes. Les critères de choix de site et les pratiques réelles de choix de site adoptées dans les différents pays semblent assez arbitraires. Ce caractère arbitraire est dû aux différentes exigences nationales, mais je pense qu'il est cependant possible d'arriver à une certaine normalisation dans les procédures de choix de site et de sécurité, les sites et les systèmes de réacteurs étant pris comme paramètres indépendants dans cet exercice.

Un site peut être défini ou classé en fonction de la libération de radioactivité acceptable en cas d'accident, ou de la charge radiologique acceptable pour le site. Ceci peut être défini sur la base des études de site et des codes de sécurité nationaux. Les systèmes de réacteurs peuvent être classés en fonction de leur capacité de confinement. L'étape suivante consiste alors à considérer la combinaison de sites donnés et la combinaison de systèmes donnés de réacteurs. On peut alors utiliser son jugement pour ajuster ou modifier les deux afin de les mettre en conformité avec le code de sécurité national.

L. CAVE (Royaume-Uni): A propos de l'utilisation du concept de l'accident maximal prévisible, je voudrais faire quelques commentaires du point de vue de celui qui fait le projet. A tout moment du développement d'un système donné de réacteur, quand une commande est sur le point d'être passée, il faut que le constructeur, l'acheteur et l'autorité qui donne les autorisations soient d'accord sur la nature des risques contre lesquels le constructeur doit fournir une protection et sur l'efficacité de cette protection. Si ceci n'est pas réalisé, il est impossible de faire les plans détaillés de façon raisonnable. En ce qui concerne les approvisionnements et les plans, il faut donc avoir une définition claire de l'accident maximal à envisager dans la conceptions; ce concept est évidemment lié à celui de l'accident maximal prévisible et c'est ce qui explique sans doute pourquoi ce dernier concept a été si largement utilisé.

L'expérience acquise sur un système donné conduira sans doute à modifier la définition de l'accident maximal à envisager dans la conception, avec des modifications correspondantes de l'équipement de secours et des changements des conditions de fonctionnement pour les réacteurs existants du même type.

V. NADEINSKI (IMCO): Je voudrais faire un bref exposé sur les activités de l'Organisation intergouvernementale consultative de la navigation maritime (IMCO). L'IMCO est une institution spécialisée des Nations Unies, qui s'occupe des problèmes maritimes internationaux, en vue de faciliter la coopération entre gouvernements dans le domaine des transports par mer. L'un de ses buts principaux est d'encourager l'adoption générale des normes les plus élevées possibles de sécurité en mer et de navigation efficace.

L'IMCO porte un vif intérêt aux problèmes d'utilisation pacifique de l'énergie atomique et, en particulier, dans les domaines de la propulsion nucléaire et du transport des substances radioactives. Elle est chargée de l'administration de la Convention internationale pour la sauvegarde de la vie humaine en mer, 1960, qui entrera en vigueur le 25 mai 1965, et, à ce sujet, a une responsabilité particulière en ce qui concerne les navires nucléaires. La Convention de 1960 pour la sauvegarde de la vie humaine en mer s'applique à tous les navires nucléaires, à l'exception des navires de guerre. Le chapitre VIII de la Convention concerne les navires nucléaires et contient un certain nombre de règles qui se rapportent surtout à des questions de principes et de procédure. Selon ces règles, les plans, la construction et les normes d'inspection, ainsi que le montage de l'installation propulsive sur un navire nucléaire, doivent être approuvés par l'autorité gouvernementale; les navires nucléaires doivent être inspectés au moins une fois par an, et doivent avoir un Certificat spécial de sécurité pour navire nucléaire, qui précise, entre autres, que le navire, étant à propulsion nucléaire, satisfait aux exigences du chapitre VIII de la Convention et est conforme aux Règles de sécurité approuvées pour le navire. Tous les autres chapitres de la Convention de 1960 s'appliquent également aux navires nucléaires, sous réserve des modifications que peut entraîner l'application du chapitre VIII.

La Conférence internationale pour la sauvegarde de la vie humaine en mer, de 1960, qui a préparé la Convention pour la sauvegarde de la vie humaine en mer, a adopté un certain nombre de recommandations applicables aux navires nucléaires, de façon à guider les gouvernements pour l'application des règles incluses dans la Convention, et à attirer l'attention sur les

principaux problèmes qui se posent dans l'état actuel du développement technique. Ces recommandations couvrent divers sujets, tels que la sûreté générale des navires nucléaires, les exigences générales de l'appareil de propulsion nucléaire, la protection et le confinement du réacteur, la protection et la sûreté radiologiques, les déchets radioactifs, le chargement du combustible et l'entretien, l'équipage, les manuels d'exploitation, l'étude de la sûreté et les contrôles spéciaux. Ces sujets seront étudiés par l'IMCO en coopération avec l'AIEA.

Un autre aspect des utilisations pacifiques de l'énergie atomique qui intéresse l'IMCO est le transport en mer de substances radioactives. L'utilisation croissante de sources radioactives pour des applications industrielles et médicales, et la construction de réacteurs dans les pays en voie de développement conduisent à un plus grand nombre de transports de substances radioactives. En application des recommandations pertinentes de la Conférence de 1960 pour la sauvegarde de la vie humaine en mer, un groupe de travail de l'IMCO prépare un code de sécurité pour le transport par mer de toutes les matières dangereuses, y compris les substances radioactives. Ce code sera éventuellement basé sur le règlement de transport récemment révisé de l'AIEA, et aussi sur les règlements maritimes existants. Ce code international contribuera à l'extension des applications pacifiques de l'énergie atomique.

Il n'y a pour le moment que deux navires marchands nucléaires en fonctionnement, le brise-glace *Lénine* et le navire pour passagers *Savannah*, mais quelques autres navires nucléaires sont en construction ou en projet. Il est possible que, dans des délais relativement courts, les navires nucléaires deviennent classiques. Au fur et à mesure que leur nombre augmentera, il faudra mettre au point des règlements internationaux plus détaillés pour les navires nucléaires. C'est l'une des tâches dont l'IMCO aura à s'occuper.

Протокол заседания 3.6

Безопасность реакторов. Размещение реакторов и удержание продуктов деления. Испытания реакторов

Председатель: Г. К. Лоуренс (Канада)

Доклад P/611 (представил С. О. В. Бергстрём)

ДИСКУССИЯ

У. Б. КОТРЕЛЛ (США): Для большинства специалистов в области ядерной безопасности, занимающихся исключительно обеспечением защиты населения от потенциальной опасности вследствие радиоактивности, которая может выделяться при работе ядерных устройств, может стать откровением заявление о том, что коэффициент безопасности как разница между принятыми максимально допустимыми уровнями нерадиоактивных загрязнений и действительными зачастую оказывается очень небольшим. Авторам доклада следует рекомендовать обратить на это внимание участников конференции. Конечно, ядерная энергия не может стать действительно перспективной в нашем обществе до тех пор, пока создаваемые ею опасности не будут оценены и сравнены с другими видами опасностей, которым подвергается общество.

Вопрос, который я хочу задать в связи с обсуждаемым докладом, касается того факта, что аварии на ядерных установках, по-видимому, не учитываются при выборе места расположения реактора. Правильно ли это? Поскольку более высокая активность будет выделяться при аварии, а не при обычной эксплуатации, то какие оправдывающие обстоятельства имеются для того, чтобы не учитывать аварии при выборе места расположения реактора?

С. О. В. БЕРГСТРЁМ (Швеция): Никаких, кроме тех, которые может дать нам продолжительный эксплуатационный опыт по наиболее важной проблеме — возможности возникновения действительно крупных аварий и их влияния на окружающую среду. Пока не накоплен такой опыт, мы должны применить свое умение и рассчитать, каким образом потенциальные опасности должны быть сопоставлены с опасностями, возникающими при нормальной эксплуатации. По сравнению с тем, что общество допускает возникновение других потенциальных опасностей и их влияния, я боюсь, что развитию применения ядерной энергии без должного основания будут препятствовать мероприятия по обеспечению безопасности, вводимые против крупных, но маловероятных аварий. Следует подчеркнуть, что понятие «нормальная эксплуатация» в нашем докладе включает такие не-

нормальные обстоятельства, как эксплуатация в условиях образования большого числа мелких отверстий в топливе, экономически допустимой утечки тяжелой воды и т. д.

Однако даже используя понятие о максимально возможных авариях в обычном смысле, технические меры обеспечения безопасности должны позволять размещать атомные электростанции там, где они должны быть установлены по обычным причинам. Как подчеркивается в нашем докладе, места расположения реакторов в Швеции до 1980 года находятся в районах, где мероприятия по предупреждению крупных ядерных аварий не составляют проблемы.

Доклад P/82 (представил Ф. де Ватэр)

ДИСКУССИЯ

Ф. К. БОЙД (Канада): Кто осуществляет анализ ядерной опасности во Франции и кому докладываются результаты этого анализа?

Ф. де ВАТЭР (Франция): Исследованием проблем безопасности реакторов занимаются различные органы, ответственные за проект, например Комиссариат по атомной энергии Франции, объединение «Электрисите де Франс» и промышленные концерны. Эти организации представляют доклады по результатам анализа проблем безопасности Подкомитету по безопасности реакторов при КАЭ Франции в Сакле. Подкомитет, председателем которого является г-н Буржуа, также контролирует осуществление мер по обеспечению безопасности действующих реакторов. Ведет эту работу группа из 15 инженеров и физиков под моим руководством.

Ф. К. БОЙД (Канада): Утверждают, что можно обеспечить защиту системы фильтрации при повышении давления в противоаварийной оболочке. Как это делается?

Ф. де ВАТЭР (Франция): Если противоаварийная оболочка оказывается открытой в момент сбрасывания давления в первом контуре, то увеличение давления в противоаварийной оболочке можно поддерживать небольшим и фильтры можно защищать от воздействия напряжений с помощью соответствующей аэродинамической системы, расположенной таким образом, чтобы уменьшить разность давления на фильтрах. Такая система рассматривается для реактора EL-4.

У. К. ДЕЙВИС (США): Вы говорите, что считаете аварию, при которой произойдет большой выброс радиоактивности, по существу невероятной. Я хотел бы знать, означает ли это, что газоохлаждаемые графитовые реакторы можно располагать в любом месте или какие-нибудь другие условия безопасности учитываются при выборе места расположения реактора?

Ф. де ВАТЭР (Франция): Французские реакторы газо-графитового типа не оборудованы системой, предохраняющей от утечки радиоактивности. Только реактор EDF-1 окружен внешней сферой, но она не служит защитной оболочкой от утечки радиоактивности, поскольку исследования безопасности показали, что это бесполезно. При авариях, включая наихудший случай разрыва трубы, температура оболочки и графита не превышает безопасных пределов, так что любая значительная эмиссия продуктов деления крайне маловероятна. Это благоприятное обстоятельство обусловлено небольшой удельной теплотворной способностью топлива и большой теплотворной способностью блока графитового замедлителя. Поэтому газо-графитовые реакторы теоретически можно размещать вблизи городов, но было бы благоразумно, по крайней мере для ближайшего будущего, подождать с осуществлением этой идеи до тех пор, пока не будет накоплен опыт эксплуатации крупных атомных электростанций, которые в настоящее время строятся на расстоянии нескольких десятков километров от больших городов.

Г. КАУТС (США): В докладе отмечено, что для будущих крупных реакторов на быстрых нейтронах можно предусматривать простую противоаварийную защиту, в частности в отношении механического влияния вероятных выбросов мощности. Может ли кто-нибудь развить эту тему?

Г. ДРЕВОН (Франция): Если принять наиболее пессимистическую точку зрения в отношении коэффициентов радиоактивности, то наиболее серьезная авария, которая, исходя из предварительного изучения работы промышленных атомных электростанций, кажется возможной, может произойти из-за относительно медленного расплавления части топлива. В связи с этим я бы рекомендовал обратиться к отчету CEA-2554. Если при определенных условиях натрий в центральной части активной зоны испарится, то может произойти значительное выделение радиоактивности. Однако результаты проведенных до сих пор исследований показывают, что скорость выделения этой радиоактивности может быть не очень большой и не вызовет взрыва.

Доклад P/182

ДИСКУССИЯ

И. ТАДМОР (Израиль): В Вашем докладе сказано, что в реакторах, окруженных противо-

аварийной оболочкой и оборудованных системой вентиляции, используются фильтры с улучшенными характеристиками. Последние результаты опытов по изучению выделения продуктов деления из реакторного топлива показывают, что отдельные частицы продуктов деления имеют размер в несколько ангстрем, а не микрон. Поэтому я хотел бы спросить, сделано ли что-нибудь для повышения эффективности абсолютных фильтров для частиц размером в несколько ангстрем. Насколько я знаю, абсолютные фильтры, имеющиеся в настоящее время, эффективны только для частиц размером несколько микрон.

Ф. Р. ФАРМЕР (Соединенное Королевство): Я полагаю, что в опытах, на которые Вы ссылаетесь, UO_2 нагревали до очень высоких температур — выше 2000° С. Анализ аварий в случае усовершенствованного газоохлаждаемого реактора позволит надеяться, что эта температура не будет достигнута. Мы нагревали облученную UO_2 до 1600—2000° С и в экспериментальном оборудовании выделилось 4000 *кюри*. Это показало, что J^{131} можно удержать с эффективностью в пределах 10^3—10^4.

Б. Г. ДУБОВСКИЙ (СССР): Не могут ли авторы докладов Р/182 и Р/275 высказать свою точку зрения относительно различных требований к выбору места расположения энергетических реакторов в Соединенном Королевстве и США?

Ф. Р. ФАРМЕР (Соединенное Королевство): Фактически не существует принципиальной разницы между требованиями безопасности в Соединенном Королевстве и в США. Различия имеются в конструктивных решениях, обусловливаемых выбором типа реакторов. Если в Соединенном Королевстве выбрана водяная система, то на ранней стадии разработки требования безопасности очень близки к таким требованиям, принятым в США. В ходе разработки появляются некоторые различия и это признак правильного развития.

К. К. БЕК (США): Я считаю, что в обоих случаях различия не очень существенны, хотя вообще используются различные реакторы. В США разработаны исключительно ценные руководства из расчета максимально возможных аварий. Подробные и унифицированные руководства были разосланы сотням различных организаций, работающих в области ядерных реакторов, и конструкторам реакторов пришлось принять меры к повышению безопасности и усовершенствованию системы мер по безопасности реакторов, поскольку были уменьшены пределы безопасного удаления. В Соединенном Королевстве принят совершенной иной подход к решению этой проблемы.

Доклад Р/380 (представил В. А. Сидоренко)

ДИСКУССИЯ

Дж. ЛИБЕРМАН (США): Конструкция и эксплуатация систем контроля реактивности, описанные в Вашем докладе, представляются интересными, и критерии этих систем кажутся довольно сходными с критериями, которыми руководствуются и в других странах. Однако я хотел бы спросить, рассматриваются ли возможности возникновения других вариантов аварий на реакторах, такие, как потеря теплоносителя? А также какие меры принимаются для защиты людей в случае аварий с выбросом продуктов деления?

В. А. СИДОРЕНКО (СССР): Мой доклад посвящен правилам обеспечения ядерной безопасности, направленным на предотвращение неконтролируемых ядерных разгонов реактора. К сожалению, я не могу сделать общего обзора по другого рода аварийным ситуациям. Однако я, конечно, могу напомнить о защитных мерах, принятых на Нововоронежской атомной электростанции. Все оборудование первого контура заключено в очень плотные железобетонные боксы, облицованные изнутри сталью. В случае потери герметичности первого контура все активные продукты остаются в пределах этих помещений. Вентиляция помещений осуществляется через фильтрационную систему в вытяжную трубу, что обеспечивает допустимую концентрацию активных продуктов в окружающем воздухе.

Б. Г. ДУБОВСКИЙ (СССР): Я хотел бы добавить, что на Белоярской атомной электростанции им. И. В. Курчатова в СССР линии первичного контура реактора высокой удельной мощности и часть линий, вторичного контура, которые могут стать радиоактивными вследствие аварии, заключены в плотные вентилируемые боксы с толстыми бетонными или чугунными стенами, служащими биологической защитой. Вентилируется также центральный реакторный зал, ведется контроль воздуха непосредственно над верхней частью реактора. Имеется центральный дозиметрический пункт, с помощью которого легко локализовать район повышенной активности и определить степень загрязненности воздуха во всех близлежащих местах. Разработаны детальные планы эвакуации персонала на случай любого варианта аварии. Аналогичные меры планируются для всех действующих реакторов. Реакторы размещаются на достаточном удалении от населенных пунктов.

Ж. СТОРРЕ (Бельгия): Я хотел бы знать, позволяют ли упомянутые в Вашем докладе

правила производить загрузку топлива водяных реакторов без органов аварийной защиты, например когда достаточная подкритичность обеспечивается жидким поглотителем, таким, как борная кислота, и когда измерительная аппаратура с начала операции загрузки находится под контролем.

В. А. СИДОРЕНКО (СССР). Правила требуют, чтобы по крайней мере два органа аварийной защиты находились в положении готовности к немедленному действию, прежде чем реактор начнет выход в критическое состояние, независимо от того, приняты ли дополнительные меры предосторожности для компенсации реактивности, например раствором борной кислоты, или реактор контролируется во всей подкритической области.

Р. Ф. ДЖЕКСОН (Соединенное Королевство): В § 4 Вашего доклада о правилах обеспечения безопасности высказывается требование, что в случае аварии вследствие расплавления топлива должен существовать двойной запас между наибольшей массой расплавленного топлива, которая может образоваться в любой точке, и критической массой. Уверены ли Вы в том, что это требование выполнимо в советских проектах реакторов на быстрых нейтронах электрической мощностью 350 и 1000 *Мвт* и что любая возможность выделения ядерной энергии во время аварии вследствие расплавления топлива может оказаться неотмеченной как во время процесса плавления, так и при окончательной установке ловушки под активной зоной?

В. А. СИДОРЕНКО (СССР): Эти правила обеспечения ядерной безопасности утверждены Государственным комитетом по использованию атомной энергии СССР, и я не слышал возражений со стороны проектировщиков реактора на быстрых нейтронах. Я не знаю, каким образом проектировщики преодолеют эту трудность, но я уверен, что они справятся с этим.

Ф. ДУХАМЕЛЬ (Франция): Я хотел бы спросить авторов или кого-нибудь из их коллег, введены ли в СССР технические ограничения на расположение реакторов вблизи мест концентрации городского населения. Например, установлена ли какая-нибудь минимальная дистанция от реактора до населенного района? Я спрашиваю это потому, что советский реактор СБР-5 расположен в непосредственной близости от застроенного района.

Б. Г. ДУБОВСКИЙ (СССР): Действительно, реактор СБР-5 на быстрых нейтронах (5000 *квт*) расположен в черте города Обнинска, но это исследовательский реактор. В СССР реакторы на быстрых и тепловых нейтронах для ядерных электростанций уже строятся вблизи населенных районов и советские ученые и инженеры работают над повышением безопасности мощных энергетических реакторов, с тем чтобы они могли быть размещены вблизи существующих городов.

Доклад P/28 (представил Ф. К. Бойд)

ДИСКУССИЯ

А. БУЛЛ (Норвегия): Предвидите ли Вы какие-нибудь трудности в обеспечении безопасности при осуществлении текущего ремонта в противоаварийной оболочке?

Ф. С. БОЙД (Канада): Так, на атомной электростанции в Дуглас-Пойнте все входы в здание реактора оборудованы воздухонепроницаемыми люками, размеры которых достаточны для ведения всех обычных ремонтных работ, поэтому во время ремонта никаких разрывов противоаварийной оболочки не делается. Если необходимо подвести или удалить более громоздкое оборудование, то это считается исключительным обстоятельством, и в этих случаях принимаются специальные меры.

К. МАХМУД (ОАР): В докладе Вы утверждаете, что «...запретная зона предусматривается для защиты населения в случае какой-либо аварии...». Является ли это единственной мерой обеспечения безопасности в отношении аварии или разрабатываются специальные планы на случай различных аварий канадского реактора? Если да, то каковы основные критерии?

Ф. К. БОЙД (Канада): Если взять в качестве примера демонстрационную атомную электростанцию, то для защиты персонала и окружающего населения разработаны меры аварийной защиты. Периодически проводятся учебные тренировки по эвакуации персонала. Поддерживается постоянный контакт с полицией, ответственными работниками здравоохранения и т. д. по вопросам эвакуации населения в случае необходимости. В различных критических местах вне запретной зоны установлены дозиметры. По первому требованию высылаются дозиметрические бригады.

Доклад P/275

ДИСКУССИЯ*

Ф. С. БОЙД (Канада): В Вашем докладе отмечается, что три энергетических реактора бы-

* См. также дискуссию к докладу P/182.

ли расположены ближе к населенным центрам, чем остальные, так как «в проекте были предусмотрены специальные меры обеспечения безопасности». Не можете ли Вы рассказать о деталях этих специальных мер?

К. К. БЕК (США): Район Малибу в Лос-Анжелесе был выделен для строительства крупного энергетического реактора потому, что в проекте были предусмотрены специальные меры обеспечения безопасности: двойная противоаварийная оболочка реактора, усовершенствованная аварийная система охлаждения и другие меры. Мы считаем, что они настолько уменьшат последствия любой серьезной аварии, что некоторое уменьшение расстояния от нормативных будет оправдано.

Й. ВЕНГЛЕР (ФРГ): Какие технические меры обеспечения безопасности Вы предусматриваете на случай разрыва парового трубопровода между противоаварийной оболочкой и турбиной в водо-водяном кипящем реакторе с прямым циклом?

К. К. БЕК (США): На случай разрыва внешнего трубопровода будут перекрыты соответствующие изолирующие вентили, установленные в первичном контуре противоаварийной оболочки реактора.

В. А. СИДОРЕНКО (СССР): Подтверждает ли опыт эксплуатации американских реакторов сложившееся представление о том, что наиболее неприятная авария связана с потерей теплоносителя? Существуют и другие типы аварий, которые не являются такими неприятными с точки зрения последствий, но возникновение которых статистически более вероятно, например неконтролируемое извлечение топливных стержней.

К. К. БЕК (США): Исследование проблемы безопасности показало, что в большинстве водяных реакторов наиболее страшными авариями, против которых предусматриваются системы обеспечения безопасности, ограничивающие последствия аварии, являются аварии вследствие потери теплоносителя. Эти аварии используются также для того, чтобы вынести суждение о соответствии мест расположения реакторов. Если разработаны соответствующие меры защиты против таких максимально возможных аварий на таких реакторах, то должны быть также разработаны соответствующие меры защиты населения против менее страшных и, следовательно, более вероятных аварий. Однако в некоторых реакторах, например в реакторах на быстрых нейтронах и в первых водяных реакторах, аварии вследствие реактивности можно считать максимально возможными.

Доклады P/714 (представил Д. А. Дейвис) и P/793 (представил В. Сурио-Бао)

По этим докладам дискуссии не было.

Доклад P/378 (представил В. Г. Пологих)

ДИСКУССИЯ

А. БУЛЛ (Норвегия): В случае серьезной аварии реактора на атомном судне, включая возможный выброс радиоактивных продуктов деления, вывод судна из порта или отвод его от ближайших населенных районов имеет важное значение. Однако на ледоколе с ядерным двигателем эта операция может замедлиться или оказаться невозможной из-за ледовой обстановки. Я хотел бы спросить, какова Ваша точка зрения на эту проблему?

В. Г. ПОЛОГИХ (СССР): Атомный ледокол «Ленин» базируется в Кольском заливе, где нет льда.

В. ВИНКЕ (Нидерланды): В Вашем докладе утверждается, что эксплуатация ледокола «Ленин» не представляет радиационной опасности для населения порта. Одной из защитных мер, принимаемых при заходе судна в порты или проходе через каналы, является снижение мощности на некоторый период перед заходом в порт или перед проходом через канал. Длительность этого периода является предметом обсуждения и зависит от субъективных факторов. Некоторые называют 6,24 или 48 *ч* или даже требуют остановки реактора и сбрасывания давления.

Не можете ли Вы сказать, какие существуют принципы по этому вопросу в СССР и какие критерии приняты для обеспечения будущего ядерных судовых двигателей?

В. Г. ПОЛОГИХ (СССР): Когда атомное судно входит в порт, мощность установки всегда снижается. Необходимость снижения давления в первичном контуре зависит от конструкции установки: в некоторых типах водоохлаждаемых реакторов некоторое снижение мощности будет, вероятно, необходимым, но, конечно, полностью давление снимать нельзя.

У. М. СТЕБЛЕР (США): Статистические данные об авариях судов, приведенные в докладе P/378, включают мелкие суда вместимостью 100 регистровых брутто-тонн и не различают небольшие повреждения от полной гибели. Согласно нашим данным в Управлении морского судоходства, в 1962 году потери судов вместимостью свыше 500 регистровых брутто-тонн составили 0,5% и вместимостью

свыше 1000 регистровых брутто-тонн составили примерно 0,26%. Средняя вместимость новых американских торговых судов составляет 9000 регистровых брутто-тонн и любые атомные суда будут иметь и такую или большую вместимость. Таким образом, я считаю, что выбранные параметры необходимо значительно видоизменить в этом исследовании, прежде чем вывести какое-либо заключение относительно потенциальной опасности использования ядерной энергии в торговых судах.

В отношении грузо-пассажирского атомохода «Саванна» я хотел бы обратить внимание на три особенности конструкции судна, обеспечивающие безопасность:

1. Противоаварийная оболочка реактора отделена от корпуса.
2. Для уменьшения последствий столкновения установлена дополнительная оболочка ближе к борту.
3. Для уменьшения последствий посадки на мель и предохранения оболочки реактора предусмотрено специальное основание.

В. Г. ПОЛОГИХ (СССР): Сейчас я не могу сказать что-нибудь определенное относительно будущих ядерных установок на ледоколах, однако, несомненно, что будут внесены усовершенствования, исходя из опыта эксплуатации ледокола «Ленин».

ОБЩАЯ ДИСКУССИЯ

А. К. ГАНГУЛИ (Индия): Я хотел бы высказать следующие замечания. Критерии выбора места расположения реакторов и существующая практика этого выбора, принятая в различных странах, кажется довольно произвольной. Эта произвольность обусловлена различными национальными требованиями, но тем не менее я думаю, что можно обеспечить некоторую стандартизацию выбора места для реакторов и условий безопасности и в качестве независимых параметров принять места расположения реакторов и реакторные системы.

Место расположения реактора можно определить или классифицировать в соответствии с приемлемым выделением активности неаварийного характера или в соответствии с радиологической характеристикой места. Это можно сделать на основе изучения места и национальных правил ядерной безопасности. Реакторные системы можно затем классифицировать в соответствии с их способностью удерживать продукты деления. Третьим шагом следует считать комбинирование данных мест и реакторных систем. После этого можно изменить или приспособить эти два параметра к национальным правилам обеспечения безопасности.

Л. КЕЙВ (Соединенное Королевство): В отношении использования понятия «максимально возможной аварии» я хотел бы высказать некоторые замечания с точки зрения проектировщика. В любое данное время при разработке конкретной реакторной системы должно быть установлено согласие между проектировщиком, покупателем и лицензирующей организацией как в отношении характера неисправности, против которой проектировщик должен принимать меры защиты, так и эффективности этой защиты. До тех пор пока это не будет сделано, детальное проектирование не может продолжаться в желаемом направлении. Поэтому на стадии закупки и проектирования должно существовать ясное понимание концепции «максимальная авария, учитываемая в проекте». Очевидно, это понятие соответствует понятию максимально возможная авария и, вероятно, это объясняет, почему последнее понятие используется более широко.

По мере накопления опыта с системои понятие «максимальная авария, учитываемая в проекте» может значительно измениться вследствие вносимых изменений в аварийное оборудование или в условия эксплуатации действующих реакторов одинакового типа.

В. НАДЕИНСКИЙ (IMCO): Я хотел бы сделать несколько замечаний относительно деятельности Межправительственной консультативной организации по морским вопросам (IMCO), являющейся специализированным агентством Организации Объединенных Наций, осуществляющим международную деятельность в области морского судоходства, цель которой заключается в том, чтобы облегчать сотрудничество между правительствами при организации морских перевозок. Одна из ее основных целей заключается в том, чтобы способствовать разработке и принятию максимально допустимых стандартов на обеспечение безопасности морских перевозок и содействовать эффективному судоходству.

IMCO особо заинтересована в проблемах мирного использования атомной энергии и, в частности, в области создания ядерных судовых двигателей и перевозке радиоактивных веществ. Она ведет дела, связанные с Международной конвенцией по обеспечению безопасности жизни на море от 1960 года, которая вступит в силу 25 мая 1965 года, и в соответствии с этим несет особую ответственность в отношении судов с ядерным двигателем. Конвенция по обеспечению безопасности от 1960 года распространяется на все атомные суда, исключая военные. Глава VIII Конвенции посвящена атомным судам и содержит ряд правил, которые главным образом ограничиваются вопросами принципа и процедуры. В соответствии с этими правилами проектирование, строительство и контроль, так же как и монтаж реакторной установки на атомном судне, должны быть санкционированы правительственным органом. Атом-

ные суда подлежат инспектированию не реже одного раза в год и должны иметь специальный «Паспорт безопасности атомного судна», который среди других положений должен удостоверять, что судно, имеющее ядерную силовую установку, подчиняется всем требованиям главы VIII Конвенции и руководствуется Правилами обеспечения безопасности, утвержденными для этого судна. Все остальные главы Конвенции 1960 года применимы также к атомным судам, за исключением упомянутых в главе VIII.

Международная конференция по обеспечению безопасности жизни на море, состоявшаяся в 1960 году, подготовила конвенцию, утвердила ряд рекомендаций в отношении атомных судов, которые явились руководством для правительств при осуществлении правил, записанных в Конвенции по обеспечению ядерной безопасности, и обратила внимание на основные проблемы, которые существуют на современном этапе технического развития. Эти рекомендации охватывают различные положения, например общая безопасность атомных судов, общие требования к установке с ядерным двигателем, защита реакторной установки, радиационная безопасность, радиоактивные отходы, загрузка топлива и обслуживание, обслуживающий персонал, руководства по эксплуатации, оценка безопасности и специальный контроль. Эти положения должны быть рассмотрены IMCO совместно с МАГАТЭ.

Другим вопросом мирного использования атомной энергии, которым занимается IMCO, являются морские перевозки радиоактивных веществ. Возрастающее использование радиоактивных источников для промышленных и медицинских целей и строительство реакторов в развивающихся странах привели к увеличению перевозок радиоактивных материалов. В соответствии с упомянутыми рекомендациями Конференции по обеспечению безопасности, состоявшейся в 1960 году, одна из рабочих групп IMCO разрабатывает свод правил по обеспечению безопасности при транспортировке по морю всех опасных продуктов, включая радиоактивные вещества. Очевидно, свод правил будет опираться на недавно пересмотренные МАГАТЭ правила транспортировки, а также на существующие морские правила. Этот международный свод правил явится вкладом в расширение мирного использования атомной энергии.

В настоящее время находятся в эксплуатации только два коммерческих атомных судна — ледокол «Ленин» и грузо-пассажирское судно «Саванна», но в стадии строительства и проектирования находятся другие атомные суда. За сравнительно короткое время атомные суда станут обычными, что будет большим достижением. По мере увеличения количества атомных судов будет расти потребность в разработке более подробных международных правил для атомных судов. Это одна из задач, стоящих перед IMCO.

Acta de la sesión 3.6

Seguridad de los reactores; emplazamiento y confinamiento; ensayos

Presidente: G. C. Laurence (Canadá)

Documento P/611 (presentado por S. O. W. Bergström)

DISCUSIÓN

W.B. COTTRELL (Estados Unidos de América): A la mayoría de los especialistas en seguridad, dedicados exclusivamente a la protección del público contra los riesgos potenciales de la radiactividad que podría liberarse durante la explotación de una instalación nuclear, les extrañará el que con frecuencia sea tan pequeño el margen de seguridad entre las concentraciones reales de contaminantes no radiactivos y los límites máximos aceptados. Ha de agradecerse a los autores de la memoria el haber presentado dicha información en esta Conferencia. En verdad, nuestra sociedad no podrá considerar la energía nuclear en su perspectiva propia, hasta que sus riesgos se evalúen sobre la misma base que se evalúan otros riesgos a los que se ve expuesta la sociedad.

La pregunta que tengo sobre esta memoria se refiere a que, al menos en apariencia, no se considera a los accidentes nucleares en la evaluación de emplazamientos para reactores. ¿Estoy en lo cierto? Puesto que en caso de accidente se libera más radiactividad que en régimen de explotación normal, ¿qué razón justifica no considerar los accidentes en la evaluación de emplazamientos?

S. O. W. BERGSTRÖM (Suecia): Nada que no sea una larga experiencia de explotación puede darnos información sobre el problema más importante:

la probabilidad real de accidentes serios y sus efectos sobre el medio ambiente. Hasta adquirir dicha experiencia, hemos de utilizar nuestro sentido común para comparar los riesgos potenciales con los creados durante la explotación normal. Al comparar cómo acepta la sociedad otros serios riesgos potenciales y los daños de ellos derivados, temo que la energía nuclear se vea obstaculizada innecesariamente por las salvaguardias contra accidentes graves pero muy poco probables. Se ha de destacar que la expresión « explotación normal », utilizada en nuestra memoria, incluye situaciones anormales, tales como: el funcionamiento con un número más bien elevado de vainas perforadas, pérdidas de agua pesada aceptables desde el punto de vista económico, etc.

Sin embargo, incluso utilizando el concepto de accidente máximo previsible en su sentido habitual, las salvaguardias técnicas permitirían la ubicación de centrales nucleares en los emplazamientos idóneos desde el punto de vista convencional. Como se pone de manifiesto en nuestra memoria, en Suecia, hasta 1980, los emplazamientos de interés de hallarán en zonas en las que no constituye un problema la protección contra accidentes nucleares graves.

Documento P/82 (presentado por F. de Vathaire)

DISCUSIÓN

F. C. BOYD (Canadá): ¿Quiénes llevan a cabo en Francia los análisis de riesgos y de quién dependen?

F. DE VATHAIRE (Francia): Las distintas organizaciones responsables de los proyectos son quienes llevan a cabo los análisis de seguridad de los reactores, es decir, el Commissariat à l'énergie atomique, Electricité de France y las empresas industriales. Dichas organizaciones someten sus informes de seguridad al análisis de un Subcomité de Seguridad de Reactores, que pertenece al Commissariat y que reside en Saclay. Dicho Subcomité, presidido por el Sr. Bourgeois, comprueba también la seguridad de los reactores en explotación. Un grupo de 15 ingenieros y físicos, bajo mi dirección, lleva a cabo este trabajo.

F. C. BOYD (Canadá): Se dice que es posible proteger el sistema de filtración en el caso de que aumente la presión en la contención. ¿Cómo se consigue esto?

F. DE VATHAIRE (Francia): Si en el momento de salida del gas del sistema primario la cubierta de contención permaneciese abierta, el aumento de presión de ésta podría ser pequeño y podrían evitarse las tensiones en los filtros utilizando un sistema aerodinámico, dispuesto de modo que se redujese la diferencia de presión a través de los filtros. Se está considerando utilizar dicho sistema en el EL4.

W. K. DAVIS (Estados Unidos de América): Vd. ha dicho que considera inverosímil el que tenga lugar un accidente con escape de grandes cantidades de radiactividad. Me pregunto si dicha afirmación quiere decir que los reactores de grafito, refrigerados por gas, pueden ubicarse en cualquier sitio, o si al evaluar el emplazamiento hay que tener en cuenta otros factores de seguridad.

F. DE VATHAIRE (Francia): Los reactores franceses de grafito, refrigerados por gas, no disponen de una cubierta de contención hermética. Solamente el reactor EDF1 está encerrado en una cubierta esférica, pero ésta no se utiliza como contención hermética, ya que los estudios de seguridad han demostrado que ello es innecesario. Los análisis han demostrado que, aún en el caso de accidentes que lleven consigo las mayores rupturas posibles de las tuberías, las temperaturas de las vainas y del grafito no superan los límites de seguridad, así que es muy poco probable el que tenga lugar un escape importante de productos de fisión; esta favorable circunstancia se debe a la baja potencia específica del combustible y a la elevada capacidad calorífica del bloque de grafito, que actúa de moderador. Por consiguiente, los reactores de grafito-gas pueden, teóricamente, ubicarse cerca de ciudades, pero sería prudente, por el momento, esperar hasta que se adquiera experiencia con las grandes centrales nucleares que ahora se construyen a varias decenas de kilómetros de las grandes ciudades.

H. KOUTS (Estados Unidos de América): Se dice en la memoria que el sistema de contención de los grandes reactores rápidos del futuro puede ser simple, en especial en cuanto a los efectos mecánicos de embalamientos verosímiles de la potencia. ¿Podría alguien extenderse más sobre este tema?

G. DREVON (Francia): Aún considerando los coeficientes de reactividad más pesimistas, el accidente más serio, según análisis preliminares de centrales comerciales, parece que sería la fusión, relativamente lenta, de parte del núcleo. En este sentido le refiero a Vd. al informe CEA-2554. Si, en ciertas condiciones, se vaporizase el sodio de la región central del núcleo, podría introducirse una cantidad considerable de reactividad. Sin embargo, de los estudios llevados a cabo hasta la fecha, parece que el ritmo de introducción de dicha reactividad no es suficiente para provocar la explosión.

Documento P/182

DISCUSIÓN

J. TADMOR (Israel): Su memoria indica que se han mejorado las características de los filtros utilizados tanto en reactores con sistema de contención cerrado como ventilado. Los experimentos recientes sobre la emisión de los productos de fisión contenidos en los combustibles nucleares han puesto de manifiesto que algunas de las partículas desprendidas tienen tamaños del orden del Ångström más bien que del orden de

la micra. En consecuencia me gustaría saber qué se ha hecho para aumentar la eficacia de los filtros absolutos que tengan que retener partículas con tamaños del orden del Ångström. Según mi información, los filtros absolutos de que se dispone en el mercado son eficaces solamente para el caso de partículas con tamaños del orden de la micra.

F. R. FARMER (Reino Unido): Creo que, en los experimentos a los que Vd. se ha referido, se calentó UO_2 a temperaturas muy altas, superiores a 2 000 °C. El análisis de accidentes del AGR nos hace confiar en que no se alcanzará esta temperatura. En una instalación experimental hemos calentado UO_2 irradiado hasta temperaturas comprendidas entre 1 600 y 2 000 °C y se han desprendido 4 000 curios. Estas experiencias han demostrado que la eficacia de retención del ^{131}I se encuentra entre 10^3 y 10^4.

B. G. DUBOVSKY (URSS): ¿Podrían los autores de los documentos P/182 y P/275 expresar su opinión sobre los distintos requisitos para la ubicación de reactores de potencia exigidos en el Reino Unido y en los Estados Unidos?

F. R. FARMER (Reino Unido): En realidad no existe una diferencia fundamental entre los principios de seguridad que se han formulado en el Reino Unido y en los Estados Unidos. Hay diferencias en las soluciones de proyecto, debido a la elección de reactores diferentes. Si el Reino Unido hubiese elegido reactores de agua, las características de seguridad en el desarrollo inicial hubiesen sido muy parecidas a las adoptadas en los Estados Unidos. Durante el desarrollo ulterior habrían aparecido algunas diferencias, lo que es un signo saludable.

C. K. BECK (Estados Unidos de América): Creo que las conclusiones a las que se ha llegado en ambos casos no son muy diferentes, a pesar de que los reactores han sido por lo general muy distintos. En los Estados Unidos hemos desarrollado guías muy valiosas basadas en el accidente máximo verosímil. Se ha orientado de forma apropiada y uniforme a cientos de organizaciones diferentes interesadas en reactores nucleares, y los proyectistas han tenido el incentivo de mejorar la seguridad y los sistemas de salvaguardias, viendo cómo se han reducido las distancias de seguridad. En el Reino Unido se ha progresado de modo completamente distinto.

Documento P/380 (presentado por V. A. Sidorenko)

DISCUSIÓN

J. LIEBERMAN (Estados Unidos de América): Son interesantes el diseño y el funcionamiento de los mecanismos de control de la reactividad descritos en su memoria, y los criterios en los que se basan parece que son muy similares a los de otros países. Sin embargo, me gustaría saber si se ha considerado la posibilidad de que tengan lugar otros tipos de accidentes, tales como la pérdida de refrigerante. Además, ¿qué medidas se han tomado para proteger al público en caso de accidentes que entrañen emisión de productos de fisión?

V.A. SIDORENKO (URSS): Mi memoria se refiere a las normas de seguridad para evitar aumentos incontrolados de la potencia del reactor y, por desgracia, no puedo dar una visión general de lo que sucedería en caso de situaciones peligrosas creadas por otras causas. Sin embargo, podría tal vez mencionar las medidas de protección puestas en práctica en la central nuclear de Novovoronezh. Todo el equipo del circuito primario se halla encerrado en celdas de hormigón armado y con recubrimiento interior de acero. En caso de fugas en el circuito primario, todos los productos activos permanecen dentro de estas celdas. Estos sistemas de contención van también provistos de un sistema de filtración instalado en el conducto de salida, lo que asegura que el aire de los alrededores no se contamine por encima de los niveles máximos permitidos.

B. G. DUBOVSKY (URSS): Me gustaría añadir que en la central nuclear I. V. Kurchatov, en Beloyarsk, URSS, las tuberías del circuito primario de este reactor de elevada potencia específica y parte de las tuberías del circuito secundario (que podría ser radiactivo en caso de accidente) se encierran también en cavidades cerradas, provistas de ventilación, y con gruesas paredes de hormigón o de fundición, que constituyen también el blindaje biológico. También se ventila la nave central del reactor y se controla el aire directamente por encima de la parte superior del reactor. Existe también un panel centralizado de vigilancia radiológica, desde el que es fácil localizar cualquier zona en la que aumente la radiactividad, y determinar la contaminación del aire en todos los compartimentos asociados. Se han elaborado planes detallados para evacuar al personal en caso de accidentes de tipos distintos. Se han planeado medidas similares para cada uno de los reactores en explotación. Los reactores se ubican a distancias razonables de los centros de población.

J. STORRER (Bélgica): Me gustaría saber si las normas mencionadas en su memoria permiten la carga del combustible en los reactores de agua sin « elementos de seguridad », por ejemplo, cuando se dispone de un grado suficiente de subcriticidad por utilizar venenos solubles, como el ácido bórico, y cuando se observan de forma continua los instrumentos de medida desde el comienzo de las operaciones de carga.

V. A. SIDORENKO (URSS): Las normas exigen que, antes de que el reactor alcance el estado crítico, al menos dos « elementos de seguridad » se encuentren en posición listos para funcionar, con independencia de que existan otros medios adicionales para com-

pensar la reactividad, en forma de soluciones de ácido bórico, o que el reactor esté controlado en todo el intervalo subcrítico.

R. F. JACKSON (Reino Unido): El párrafo 4 de las normas de seguridad, descritas en su memoria, parece requerir, que, durante un accidente de fusión, habrá que disponer de un factor de seguridad de 2 entre la masa crítica y la mayor cantidad de combustible fundido que se pueda acumular en cualquier punto. ¿Cree Vd. que se puede satisfacer este requisito al diseñar los reactores rápidos propuestos en la Unión Soviética de 350 y 1 000 MW(e), y que se puede descartar cualquier posibilidad de que se libere energía nuclear, tanto durante el proceso de la fusión como en el cubículo de recogida final por debajo del núcleo?

V. A. SIDORENKO (URSS): Las normas de seguridad merecieron la aprobación de la Comisión Estatal de la Energía Atómica, y no he oído que los proyectistas de reactores rápidos hayan formulado ninguna objeción. No sé cómo resolverán los proyectistas esta dificultad, pero creo que lo lograrán.

F. DUHAMEL (Francia): Me gustaría preguntar a los autores, o a cualquiera de sus colegas, si en la URSS se imponen restricciones técnicas a los reactores que se instalen cerca de ciudades. Por ejemplo, ¿se ha prescrito alguna distancia mínima entre el reactor y las zonas urbanas? Hago esta pregunta porque el reactor soviético SBR-5 se encuentra cerca de una zona urbana.

B. G. DUBOVSKY (URSS): El reactor rápido SBR-5 de 5 000 kW se encuentra efectivamente ubicado dentro del cascó urbano de la ciudad de Obninsk, pero sa trata, en este caso, de un reactor de investigación. En la URSS, los reactores rápidos y térmicos de las centrales nucleares se están ya construyendo cerca de zonas urbanas y los científicos e ingenieros soviéticos están trabajando sobre la manera de aumentar la seguridad de reactores de potencia aún mayores, de modo que puedan ubicarse cerca de ciudades.

Documento P/28 (presentado por F. C. Boyd)

DISCUSIÓN

A. BULL (Noruega): ¿Prevén Vds. dificultades relacionadas con la seguridad del sistema de contención, al llevar a cabo operaciones de mantenimiento?

F. C. BOYD (Canadá): En el caso de la central nuclear de Douglas Point, por ejemplo, todas las entradas al edificio de contención disponen de exclusas de aire, que tienen tamaño suficiente para llevar a cabo todas las operaciones normales de mantenimiento, de modo que no hay que abrir la contención a causa del mantenimiento. Si hubiese que introducir o retirar equipo más grande, se consideraría como situación excepcional y se prepararían procedimientos especiales.

K. MAHMOUD (República Arabe Unida): En su memoria Vd. afirma que « ... se pretende que la zona de exclusión sirva para proteger al público en caso de accidente... ». ¿Es esta la única precaución con respecto a un accidente, o hay que estudiar planes especiales de emergencia aplicables a los distintos tipos de reactores canadienses? En caso afirmativo, ¿cuáles son los criterios más importantes a considerar?

F. C. BOYD (Canadá): Tomando como ejemplo la central nuclear NPD, se han preparado normas especiales para proteger al personal de la central y a la población vecina. Periódicamente se llevan a cabo simulacros de evacuación. Se mantienen contactos regulares con la policía, autoridades sanitarias, etc., con el fin de preparar la evacuación del público, si fuere necesario. Fuera de la zona de exclusión se instalan detectores de radiactividad en los lugares más importantes. Se puede disponer también con prontitud de equipos de vigilantes radiológicos.

Documento P/275

DISCUSIÓN *

F. C. BOYD (Canadá): Afirma Vd. en su memoria que se han instalado tres reactores de potencia más cerca de centros de población de lo que se hubiesen permitido porque « en el proyecto se incorporaron ciertos sistemas específicos de seguridad ». ¿Podría Vd. dar detalles sobre dichos sistemas específicos?

C. K. BECK (Estados Unidos de América): En el emplazamiento de Malibu, en la ciudad de Los Angeles, se aprobó la instalación de un reactor de potencia elevada a causa de ciertas salvaguardias incorporadas al proyecto: sistema de contención doble, sistemas de refrigeración de emergencia mejores y otros sistemas de seguridad. Creemos que estos sistemas reducirían tanto las consecuencias de cualquier accidente serio, que es posible justificar la reducción de los requisitos de distancia.

J. WENGLER (República Federal de Alemania): ¿Qué tipo de salvaguardia técnica propondría Vd. para contrarrestar los efectos de la rotura de una conducción de vapor entre la cubierta de contención y la turbina, en el caso de un reactor de agua en ebullición de ciclo directo?

C. K. BECK (Estados Unidos de América): En caso de rotura de las conducciones exteriores, se aislaría el circuito mediante, válvulas apropiadas, instaladas en las tuberías del primario junto a la pared de contención.

V. A. SIDORENKO (URSS): De la experiencia de explotación de los reactores de los Estados Unidos, ¿se puede sostener la idea de que el peor tipo de accidente es la pérdida de refrigerante? ?Podrían tener lugar otros tipos de accidentes, no tan malos desde el

* Véase también la discusión del documento P/182.

punto de vista del daño producido, pero más frecuentes desde el punto de vista estadístico, como por ejemplo, la extracción incontrolada de una barra de control?

C. K. BECK (Estados Unidos de América): Las investigaciones sobre seguridad han demostrado que, en la mayoría de los reactores de agua, los accidentes peores, contra los que hay que instalar sistemas de seguridad del tipo limitativo de consecuencias, son los causados por la pérdida de refrigerante. Dichos accidentes se utilizan también para juzgar lo adecuado de los emplazamientos. Si en tales reactores existe protección adecuada contra estos accidentes máximos probables, existirá también la adecuada protección del público en caso de accidentes menos graves, tal vez más probables. Sin embargo, en algunos reactores, por ejemplo, en los reactores rápidos y, al principio, en los reactores de agua, los accidentes de reactividad pueden considerarse como los máximos probables.

Documentos P/714 (presentado por D. A. Davies) y P/793 (presentado por V. Surya-Rao)

No hubo discusión de estas memorias.

Documento P/378 (presentado por B. G. Pologikh)

DISCUSIÓN

A. BULL (Noruega): En caso de accidente serio en el reactor de un buque nuclear, del que pueda resultar la emisión de productos de fisión radiactivos, es importante retirar el buque del puerto o de zonas pobladas próximas. Sin embargo, en el caso de un rompehielos propulsado por energía nuclear, dicha operación puede sufrir retrasos o ser imposible a causa de las condiciones del hielo. Me gustaría escuchar sus puntos de vista sobre este problema.

B. G. POLOGIKH (URSS): La base del rompehielos *Lenin* se encuentra en la bahía de Kola, donde no hay hielo.

W. VINKE (Países Bajos): En su memoria, afirma Vd. que la explotación del rompehielos *Lenin* no supone riesgos radiactivos para la población de los puertos. Una de las medidas de protección que se pueden tomar cuando el barco esté en puerto o pase por un canal es reducir su potencia durante un cierto período de tiempo antes de la entrada. La cuantía de este período es tema de discusión y depende del margen de seguridad que se quiera adoptar. Se podría estipular, digamos, 6, 24 ó 48 horas o incluso el que el reactor deba ser apagado y su presión reducida.

¿Podría Vd. dar alguna indicación del punto de vista soviético sobre este tema, y qué criterios se han adoptado para garantizar el futuro de la propulsión nuclear de buques?

B. G. POLOGIKH (URSS): Cuando un buque nuclear se acerca a puerto, siempre se reduce la potencia del reactor. La necesidad de reducir la presión del circuito primario depende del equipo; en ciertos tipos de reactores refrigerados por agua probablemente se requerirá la reducción de presión, pero, naturalmente, no es necesario reducir la presión del todo.

U. M. STAEBLER (Estados Unidos de América): Los datos estadísticos utilizados en el documento P/378 sobre porcentajes de accidentes en barcos incluyen unidades pequeñas, hasta de 100 toneladas de registro bruto, y no se distinguen adecuadamente los pequeños daños de las pérdidas totales. Según los datos de nuestra Administración Marítima, en 1962, el porcentaje de pérdidas de buques de más de 500 toneladas de registro bruto fue de 0,5%, y de 0,26% en el caso de buques de más de 1 000 toneladas de registro bruto. El tamaño medio de los modernos mercantes americanos es de unas 9 000 toneladas de registro bruto y cualquier buque nuclear sería de ese tamaño o mayor. Por consiguiente, sugiero que en este estudio se deben modificar sustancialmente los parámetros elegidos, antes de emitir cualquier juicio sobre los riesgos potenciales que acarrea la utilización de la energía nuclear en los barcos mercantes.

Con respecto al buque nuclear NS *Savannah*, deseo llamar su atención sobre tres de las características especiales de seguridad de su diseño:

a) Una vasija de contención separada de la estructura del casco;

b) Una estructura adicional exterior a la contención para hacer mínimos los efectos de las colisiones;

c) Un soporte proyectado para reducir al mínimo los efectos en caso de encallar y para proteger a la vasija de contención y a su contenido cuando el barco se encuentre en posiciones extremas.

B. G. POLOGIKH (URSS): Por el momento, nada puedo decir definitivo sobre el futuro de la energía nuclear en los rompehielos, pero no cabe duda de que se introducirán mejoras a la vista de la experiencia de explotación obtenida con el rompehielos *Lenin*.

DISCUSIÓN GENERAL

A. K. GANGULY (India): Me gustaría hacer los siguientes comentarios. Los criterios y prácticas actuales para evaluar emplazamientos adoptados en los distintos países parecen más bien arbitrarios. Esta arbitrariedad se debe a los diferentes requisitos nacionales, pero, sin embargo, creo es posible establecer normas en los procedimientos de seguridad y en la evaluación de emplazamientos; en estos comentarios emplazamientos y reactores se consideran como parámetros independientes.

Se puede definir o clasificar un emplazamiento según sean aceptables la emisión en caso de emergencia o la carga radiactiva que pueda soportar. Esto puede hacerse tomando como base el estudio del emplaza-

miento y las normas nacionales de seguridad. Los distintos tipos de reactores pueden entonces clasificarse atendiendo a su sistema de contención. El tercer paso sería considerar la combinación de emplazamientos y reactores. Con sentido común se pueden entonces ajustar o modificar ambos para que se atengan a las normas de seguridad nacionales.

L. CAVE (Reino Unido): Con respecto al suo del concepto de accidente máximo previsible, me gustaría comentarlo desde el punto de vista del proyectista. En cualquier momento del desarrollo de un tipo de reactor, cuando se va a hacer un pedido, el proyectista, el comprador y las autoridades que emitan las licencias deben estar de acuerdo en lo que respecta a la naturaleza de las averías contra las que el proyectista debe proteger al sistema, así como con respecto a la efectividad de tal protección. Sin esto, el proyecto detallado no puede proseguir de forma razonable. En la etapa de suministro y diseño es necesario, por consiguiente, el haber definido con claridad « el accidente máximo a considerar en el proyecto »; sin duda, este concepto está relacionado con el concepto de accidente máximo probable, lo que explica probablemente el porqué se ha utilizado con tanta frecuencia el último concepto.

A medida que se adquiere experiencia con el sistema en desarrollo, podría muy bien cambiar el « accidente máximo a considerar en el proyecto », lo que da lugar a modificaciones en el equipo de emergencia o a cambios en las condiciones de explotación de los reactores del mismo tipo ya instalados.

V. NADEINSKI (OCMI): Me gustaría exponer el trabajo llevado a cabo por la Organización Consultiva Marítima Intergubernamental. La OCMI es un organismo especializado de las Naciones Unidas que se encarga de los asuntos marítimos internacionales, y cuyo objetivo es facilitar la cooperación entre gobiernos en relación con la navegación marítima. Uno de sus objetivos fundamentales es estimular la adopción general de las mejores normas de seguridad marítima y las de navegación más eficaces.

La OCMI tiene especial interés en los problemas de la utilización pacífica de la energía atómica y, en particular, en la propulsión nuclear y en el transporte de sustancias radiactivas. Administra la Convención internacional para la seguridad de la vida en el mar, de 1960, que entrará en vigor el 25 de mayo de 1965, y, en este sentido, tiene una responsabilidad especial con respecto a los buques nucleares. La Convención para la seguridad de 1960 se aplica a todos los buques nucleares, exceptuando los de guerra. El capítulo VIII de la Convención se refiere a los buques nucleares y contiene unas cuantas normas que, en su mayor parte, se refieren a cuestiones de principio y de procedimiento. Según esa reglamentación, el proyecto, construcción y normas para la inspección, así como el montaje del reactor de un buque nuclear, deben ser aprobadas por la autoridad gubernamental; los buques nucleares deben ser inspeccionados, al menos una vez al año, y deberán llevar un Certificado de seguridad de buque nuclear especial que, entre otras cosas, establecerá que el barco, por ser nuclear, cumple todos los requisitos estipulados en el capítulo VIII de la Convención, y se ajusta al informe de seguridad que sirvió para aprobar la explotación del barco. El resto de los capítulos de la Convención de 1960 se aplica también a los buques nucleares, salvo que se haga constar otra cosa en el capítulo VIII.

La Conferencia internacional sobre la seguridad de la vida en el mar de 1960, que preparó la Convención para la seguridad, aprobó una serie de recomendaciones aplicables a los buques nucleares, con objeto de asesorar a los gobiernos en la aplicación de las normas contenidas en la Convención, y hacer resaltar los principales problemas que existen en la etapa actual del desarrollo técnico. Dichas recomendaciones abarcan varios aspectos, tales como la seguridad general de los buques nucleares, los requisitos generales del reactor propulsor, la protección y el confinamiento del reactor, blindaje y seguridad contra las radiaciones, residuos radiactivos, carga de combustible y mantenimiento, selección de personal, manuales de explotación, análisis de seguridad y control especial. Estos temas han de ser tratados por la OCMI en cooperación con el OIEA.

Otro aspecto de la utilización pacífica de la energía atómica, también relacionada con la OCMI, se refiere al transporte por mar de sustancias radiactivas. La creciente utilización en la medicina y en la industria de fuentes radiactivas y la construcción de reactores en países en vías de desarrollo hacen que aumenten los envíos de material radiactivo. De acuerdo con la importante recomendación de la Conferencia sobre la seguridad celebrada en 1960, uno de los grupos de trabajo de la OCMI está elaborando un reglamento de seguridad para el transporte por mar de todas las mercancías peligrosas, incluyendo las sustancias radiactivas. El reglamento se basará en las normas, revisadas hace poco, del OIEA sobre transporte y también en la reglamentación marítima ya existente. Este reglamento internacional contribuirá a extender la utilización pacífica de la energía atómica.

Hasta ahora solamente hay en explotación dos barcos mercantes propulsados por energía nuclear: el rompehielos *Lenin* y el barco de pasajeros *Savannah*, pero se encuentran en construcción o en fase de proyecto algunos otros barcos nucleares. Podría suceder que en un período de tiempo relativamente corto los barcos nucleares llegaran a ser de uso común. A medida que aumenta el número de buques nucleares crece la necesidad de elaborar normas internacionales más detalladas referentes a los barcos nucleares. Esta es una de las tareas con la que se enfrenta la OCMI.

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