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FOREWORD

The Fourth International Conference on the Peaceful Uses of Atomic Energy, held at Geneva from 6 to 16 September 1971 under the Presidency of Glenn T. Seaborg, was jointly sponsored by the United Nations and the International Atomic Energy Agency. The conference sessions were held at the Palais des Nations. During the same period a Governmental Scientific Exhibition on the theme "Atoms for Development" was displayed at the Palais des Expositions in Geneva.

The Proceedings are published in 15 main volumes, fourteen of which contain all the 514 papers presented at the conference. The papers are printed in English, French, Russian or Spanish, and the abstracts in all four languages; the discussions are in English. The fifteenth volume contains a Subject Index, an Author Index (including discussion contributors), a Paper Number Index, a complete Contents List of all volumes, and a List of Delegations. There are three supplementary volumes containing the discussions in French, Russian and Spanish respectively.

The conference, which attracted more than four thousand participants, observers and journalists, was planned to interest not only scientists and technologists but also public officials, economists and planners. It thus had a somewhat broader scope than the conferences of 1955, 1958 and 1964. The main topics were grouped under the following six headings: nuclear power; nuclear fuels and materials; health, safety and legal aspects; isotopes and irradiation; international and administrative aspects; and selected subjects of particular interest to developing countries.

The fourth Geneva Conference proved again to be an exceptional forum enabling those working throughout the world on the peaceful application of atomic energy to exchange the latest information on the discoveries, projects and problems of both developed and developing nations.

SUMMARY OF COMPLETE CONTENTS

- VOLUME 1 Opening and closing speeches; special talks; world energy needs and resources, and the role of nuclear energy; national and international organizations; narrative of the exhibits.
- VOLUME 2 Performance of nuclear plants; costing of nuclear plants; fuel management.
- VOLUME 3 Safety aspects of nuclear plants; legal aspects of nuclear energy.
- VOLUME 4 Integration of nuclear plants in electrical networks; integrated planning of nuclear industry; fuel materials technology.
- VOLUME 5 Breeder and advanced converter reactors.
- VOLUME 6 Small and medium power reactors; desalination and agro-industrial complexes; role of research reactors; impact of nuclear energy in developing countries.
- VOLUME 7 Advanced energy concepts; peaceful nuclear explosions; special applications, including ship propulsion; controlled thermonuclear reactions; application of transuranium isotopes.
- VOLUME 8 Uranium and thorium ore resources; fuel fabrication and reprocessing.
- VOLUME 9 Isotope enrichment; fuel cycles; safeguards.
- VOLUME 10 Effects of irradiation on fuels and materials.
- VOLUME 11 Health physics and radiation protection; radioactive waste management; the environment and public acceptance.
- VOLUME 12 Nuclear methods in food production; education and training, and public information.
- VOLUME 13 Medical applications; radiation biology.
- VOLUME 14 Applications of nuclear techniques in industry and natural resources.
- VOLUME 15 Indexes and lists.

Separate Supplementary Volumes of the Discussions in French, Russian and Spanish

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AVAILABILITY OF TRANSLATIONS

The following translations of papers in this Volume, made by the countries of origin, can be obtained from the IAEA on request:

English translation of Russian language paper A/CONF.49/P/548.

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AGENDA ITEM 1.5

Review of safety aspects of nuclear power plants

Revue des problèmes de sécurité dans les centrales nucléaires

Обзор вопросов безопасности на ядерных электростанциях

Problemas de seguridad en las centrales nucleares

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SOLUTIONS OF SAFETY PROBLEMS ENCOUNTERED IN THE OPERATION OF THE ITALIAN NUCLEAR POWER PLANTS

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Abstract-Résumé-Аннотация-Resumen

SOLUTIONS OF SAFETY PROBLEMS ENCOUNTERED IN THE OPERATION OF THE ITALIAN NUCLEAR POWER PLANTS.

The operation of the Italian nuclear power plants, which are of three different types: Latina (Magnox), Garigliano (BWR), Trino Vercellese (PWR), has allowed ENEL and CNEN to gain a considerable amount of experience in the field of nuclear plant safety. The paper deals with some of the main problems encountered during operation and with the solutions adopted. In solving these problems, including the most critical ones, the main aim was to continue plant operation while maintaining the safety margin at a level comparable with the design values. The paper deals with some modifications suggested by the technological evolution which have been introduced in the plants to improve their safety standards.

EXPERIENCE ACQUISE DANS LA SOLUTION DES PROBLEMES DE SECURITE CONCERNANT LES CENTRALES NUCLEAIRES ITALIENNES.

L'exploitation des trois centrales nucléaires italiennes, de type différent: Latina (Magnox), Garigliano (à eau bouillante), Trino Vercellese (à eau sous pression), a permis au CNEN et à l'ENEL d'acquies une expérience considérable dans le domaine de la sécurité des centrales nucléaires. Le mémoire décrit les différents problèmes qui se sont posés en cours d'exploitation, ainsi que les solutions adoptées. Tous les problèmes, y compris les plus critiques, ont été examinés et résolus de façon à permettre la poursuite de l'exploitation avec une marge de sécurité respectant les prévisions du projet. Le mémoire décrit également les modifications qui ont été apportées aux centrales, en tenant compte de l'évolution technologique, afin d'optimiser les normes de sécurité.

РЕШЕНИЕ ПРОБЛЕМ БЕЗОПАСНОСТИ НА ИТАЛЬЯНСКИХ АТОМНЫХ ЭЛЕКТРОСТАНЦИЯХ.

Эксплуатация итальянских атомных электростанций трех различных типов – в Латине (магноксовый реактор), в Гарильяно (реактор с кипящей водой) и в Трино Верчелезе (реактор с водой под давлением) – позволила Национальному комитету по атомной энергии и Национальному энергетическому управлению накопить значительный опыт в области обеспечения безопасности на атомных электростанциях. В докладе изложены различные проблемы, возникшие в ходе эксплуатации, а также принятые решения. При решении этих проблем, в том числе самых актуальных, предусматривалось продолжение эксплуатации станций с соблюдением пределов безопасности на уровне, сравнимом с уровнем таких пределов, предусмотренных проектом. В докладе описаны также модификации, осуществленные на атомных электростанциях, с учетом развития технологии, с целью улучшения параметров безопасности.

SOLUCION DE PROBLEMAS DE SEGURIDAD SUSCITADOS DURANTE LA EXPLOTACION DE LAS CENTRALES NUCLEARES ITALIANAS.

La explotación de las centrales nucleares italianas - que son de tres tipos distintos: Latina (Magnox), Garigliano (de agua hirviente), Trino Vercellese (de agua a presión) - ha dado ocasión a que ENEL y CNEN adquieran una cantidad considerable de experiencia en el campo de la seguridad de las instalaciones nucleares. Este trabajo trata de algunos de los problemas principales encontrados durante la explotación y de las soluciones que se adoptaron. Al resolver estos problemas, incluso los más críticos, se ha intentado principalmente continuar la explotación de la instalación manteniendo el margen de seguridad en un nivel

comparable con los valores proyectados. Este trabajo se refiere a algunas modificaciones, sugeridas por la evolución tecnológica, que se han introducido en las instalaciones a fin de mejorar sus niveles de seguridad.

1. INTRODUCTION

The operation of the Italian nuclear power stations, which are of three different types (pressurized-water, boiling-water and gas-cooled graphite-moderated), gave ENEL and CNEN the opportunity to gain a certain amount of experience of safety problems that may occur during operation.

In this paper we stress that, once a power plant has been licensed, not all the safety problems can be considered solved, since difficulties may arise which require the attention of the safety engineers, the operators, the designers and the licensing authorities.

Particular cases are summarized here which deal with phenomena such as vibration and corrosion to prove that, with due care, proper solutions can be found and the plants can be brought back into operation without decreasing the safety margins against failure. In addition, a few modifications or improvements are mentioned that were adopted in some systems to increase plant safety in the light of up-to-date safety analysis criteria.

The Italian nuclear power stations started commercial operation between January 1964 and December 1965 (Table I). The respective outputs are listed in Table II.

The paper briefly describes the difficulties and repairs at the three stations, namely: Garigliano station (BWR) – rupture of liquid-poison sparger; modifications to containment spray system; corrosion product deposition; control-rod drive housing support; ventilation system valves; Latina station (Magnox) – steel oxidation in CO_2 ; Trino station (PWR) – failure of internals.

2. GARIGLIANO POWER STATION

2.1. Rupture of liquid-poison sparger

2.1.1. General

During December 1965, while we were trying to plug the bottom nozzles of the pressure vessel from inside the reactor to repair the drain pipes, it was found that the liquid-poison sparger was no longer in place.

This sparger was made of two 2-in. AISI-304 stainless-steel semi-circular parts, so as to form two half rings concentric with the pressure-vessel wall. The two sections were welded to a special T-shaped piece, which was connected with the liquid-poison feeding system. Twenty-two 1/4-in. holes were arranged along the pipe generatrix for distribution of the solution within the reactor.

The liquid-poison sparger appeared to be cut off at the bracket welded to the pressure-vessel wall. The two sparger sections were lying on the bottom of the pressure vessel, broken into several fragments. It was

TABLE I. MAIN DATES FOR THE THREE POWER PLANTS

Item	Latina	Garigliano	Trino Vercellese
Beginning of construction ^a	Nov. 1958	Nov. 1959	Jul. 1961
First criticality	Dec. 1962	June 1963	Jun. 1964
First synchronization	May 1963	Jan. 1964	Oct. 1964
First power	Dec. 1963	May 1964	Dec. 1964 ^b Nov. 1965 ^c
Beginning of commercial operation ^d	Jan. 1964	May 1964	Dec. 1965
First general maintenance ^e	May 1965	Sep. 1965	-

^a Date on which the ground on the site was broken.

^b 186-MW(e) turboalternator at full power.

^c 86-MW(e) turboalternator at full power; thermal output 825 MW, corresponding to an average electric output of 252 MW(e) (equal to the maximum design thermal output for the first core). The target output of the two turboalternators is 272 MW(e).

^d The beginning of commercial operation is taken as the moment when the plant has reached nominal power and is in a position to continue operating smoothly.

^e General maintenance entails the shutdown of the whole station for several weeks.

TABLE II. GROSS OUTPUT AND PLANT CAPACITY FACTORS:
NOMINAL POWER OF THE THREE POWER PLANTS

Year	LATINA		GARIGLIANO		TRINO VERCELLESE		TOTAL
	kWhx10 ⁶	utilization	kWhx10 ⁶	utilization	kWhx10 ⁶	utilization	
1963	322						322
1964	1544	83.70%	737	78.18%	122		2403
1965	1519	82.57%	967	69.00%	1025	57.80%	3511
1966	1452	78.13%	817	59.30%	1594	72.40%	3863
1967	1586	86.21%	919	65.70%	647	29.30%	3152
1968	1544	83.70%	1033	73.50%	0		2577
1969	496	27.00%	1182	84.33%	0		1678
1970	1191	64.72%	742	52.94%	1243	56.35%	3176
Total	9654		6397		4631		20682

ascertained that all the fragments had been found and removed from the reactor by putting the pieces together as if to reform the sparger.

Several metallurgical tests were performed on a sparger fragment and on a bracket which showed a rupture near the bolt, beneath the bend, and the conclusion was reached that the liquid-poison sparger had broken in the first period of reactor operation.

2.1.2. Investigations into the damage caused by the fragments

When the rupture of the liquid-poison sparger was discovered, an investigation was carried out to establish whether the pressure vessel and its structures had been damaged as a result of the impacts of the fragments. The following items were examined: the pressure-vessel walls beneath the lower grid; the bottom of the pressure vessel; the guide-tube weldings; the control-rod guide tubes; the in-core chamber guide tubes; the lower grid and supports; the spray nozzles of the reactor emergency cooling system.

The investigation was performed with an underwater television camera designed for horizontal and vertical shots and concurrent recording on magnetic tape (AMPEX), and a borescope adjusted for horizontal and vertical observations.

The resolving power of these tools was so high that it enabled us to discover dents down to 0.3 - 0.5 mm wide.

No damage to the pressure vessel or its internal structures that would affect reactor safety was observed. The impact of the fragments on ductile material had produced dents on the curved surfaces, which are more resistant, and cuts on the sharp edges. In some areas, scratches had been caused where the rubbing and cutting action of the fragments had continued for a long time. All the damaged areas were easily identified.

2.1.3. Effects of the removal of the liquid-poison sparger on reactor operation

After the sparger removal, the only means of injecting the liquid poison into the pressure vessel was the inlet nozzle. To mix the sodium pentaborate solution with the core cooling water, there was only the turbulence of the recirculation water jet.

It might be mentioned here that the liquid-poison sparger had not been conceived as a fast, automatic backup for the normal reactor scram system, but as a device that the operator could use at his discretion after having ascertained that the reactor could not be kept sub-critical in cold conditions by any other means.

When the recirculating pumps are in operation, the water turbulence at the core inlet ensures that the sodium pentaborate mixes with the water flowing into the core. If the reactor is at power in natural circulating conditions, there will still be water turbulence from the strong water flow in this condition, as demonstrated in the startup tests.

2.1.4. Conclusions

On the basis of our findings, we reached the conclusion that the rupture of the liquid-poison sparger and the long period that the fragments had remained on the bottom of the pressure vessel had in no way affected safety conditions. The plant was therefore brought into operation again, but it was decided during the refuelling shutdown that other inspections should be made in the reactor pressure vessel.

Two inspections have been carried out to date and no deterioration of the situation has been observed. The main purpose of the inspections was to check all the internal components that had been affected by the rupture

of the liquid-poison sparger ring, together with some other areas, in such a way that all the internal components would receive a general inspection over a period of three to four years.

2.2. Modifications to the containment spray system

2.2.1. General

To improve the characteristics of the containment system, some modifications were made to the internal spray system of the containment. The original system distributed and atomized the water to decrease the overpressure resulting from a reactor primary system rupture. The new system, which sprays a solution of sodium thiosulphate, also removes the iodine released in the containment sphere.

2.2.2. Modifications

The following modifications were made (Fig.1): addition of a tank for the preparation, dosage and storage of the sodium thiosulphate solution; installation of two metering pumps; modification of the number, design and layout of the sparging nozzles.

The new nozzles have the following characteristics. The nozzle worth is 7/16 in. to avoid accidental clogging; the water is not atomized through centrifugation but through impact of the water stream against a conveniently serrated baffle, and, as a result, the drop distribution is improved over the entire volume instead of being in the form of the conical distribution made by the supply nozzles.

The design of the baffle ensures that, when a drop detaches, it is not greater than 1 mm in diameter. The high speed of the drops in the atmosphere guarantees their further break-up, so that the average size of no more than 0.7 mm assumed by the calculation is quite conservative. The nozzles and the related pipes have been arranged in such a way that they are protected against missiles in the event of an accident.

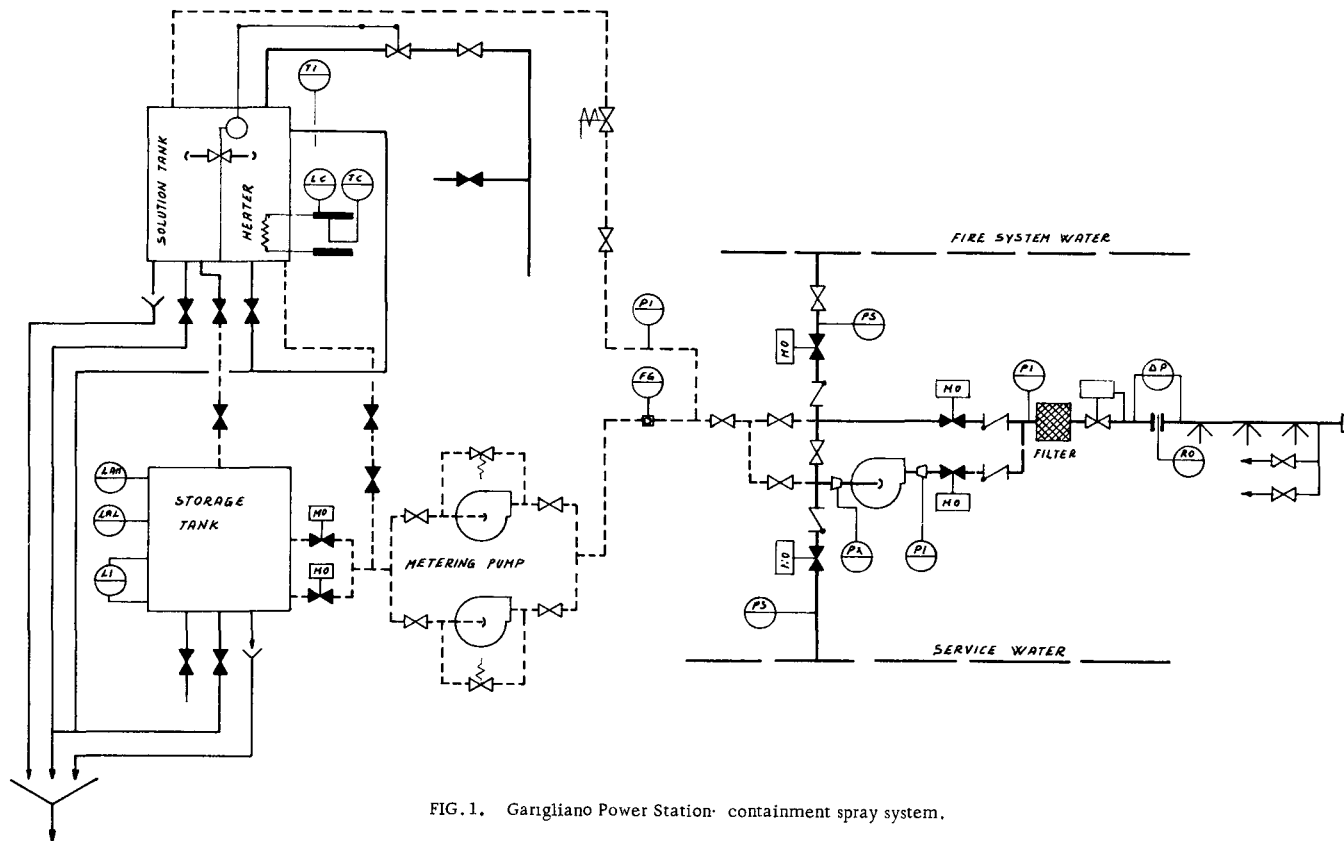
2.2.3. Periodical tests

During plant operation, the following test program is carried out on a routine basis: analysis of the thiosulphate solution; control of the performance of the motor-operated valve; pump performance; cleaning of the piping.

During the plant shutdown, the filter on the inlet line inside the containment is visually inspected and the pressure drop between the regulation valve and the nozzles is measured to ensure that the pipes are not clogged.

2.3. Prevention of corrosion product deposition

At the Garigliano station, the condensate is purified by mixed beds and returned to the main steam drum and secondary steam generators by way of two LP and two HP feedwater heaters. Water purity in the primary steam is maintained by the clean-up demineralizer, which processes 40 tons/h of primary water.



During the first (1A) and second (1B) fuel cycles, large amounts of crud were deposited at the fuel assembly inlets, causing an increased core pressure drop and entailing stricter operating limitations. At the end of both cycles, these deposits had to be removed mechanically. The deposits on the bottom tie-plate, which were responsible for the abnormal increase in core Δp , consisted of a thick layer of porous black material.

Chemical and structural analyses indicated the following composition:

Copper 60% in the form of CuO and 6 CuO · Cu₂O

Iron 24% in the form of NiO · Fe₂O₃

Nickel 16% in the form of NiO · Fe₂O₃ and NiO

After measures had been taken to minimize CuO pick-up from the Cu-Ni alloy FW heaters, no appreciable increase in the core Δp was observed during Cycle IC (August 1967 to June 1968) and no fuel cleaning was necessary.

During the June 1968 shutdown, the FW heaters were replaced with AISI 304 L units, which definitely solved the problem of crud deposit at the fuel inlet.

2.4. Installation of support for the control-rod drive housings

The control-rod drive housings are welded to the bottom of the pressure vessel. In the very unlikely event of a failure in one of the welds, the pipe could be ejected from the pressure vessel, dragging out the control rod. The step of positive reactivity resulting from rod ejection causes a power excursion, which could result in undesirable values of fuel enthalpy.

In compliance with the criteria adopted for BWR's designed after the Garigliano plant, we decided to install a supporting structure beneath the pressure vessel. This structure limits the stroke of the control-rod drive housing and would also limit a power excursion.

The structure components (joists, tie-rods, springs, nuts, screws and pillars) were designed to resist, with a certain margin, rupture of the most critical control-rod housing (Figs 2 and 3).

The support was also analysed in regard to earthquake effects.

2.5. Ventilation system valves

To ensure compliance with the limitation on the daily containment leakage rate (0.1%), provision was made to maintain a water head on the fast closure isolation valves of the ventilating system.

For this purpose, a 25-mm pipe was branched off the containment spray system and connected to each of the two ventilating ducts within the container. If an accident occurs, the spray system operating initially on water will keep the containment pressure down and create a water seal on the ventilating valves.

The ventilating ducts are filled with water in ten minutes.

The thiosulphate is injected into the spray system 15 minutes after the accident, when the iodine release becomes significant. In order to avoid overloading on the ventilating ducts, an overflow has been installed to keep the water level constant at 10 cm over the ventilating ducts. The ventilating duct sections to be flooded have been designed for a pressure of 27 atm abs. and are clamped on both sides of containment penetrations.

The additional stresses on the ducts caused by flooding are not important. The total stress is below the resistance limits of the material.

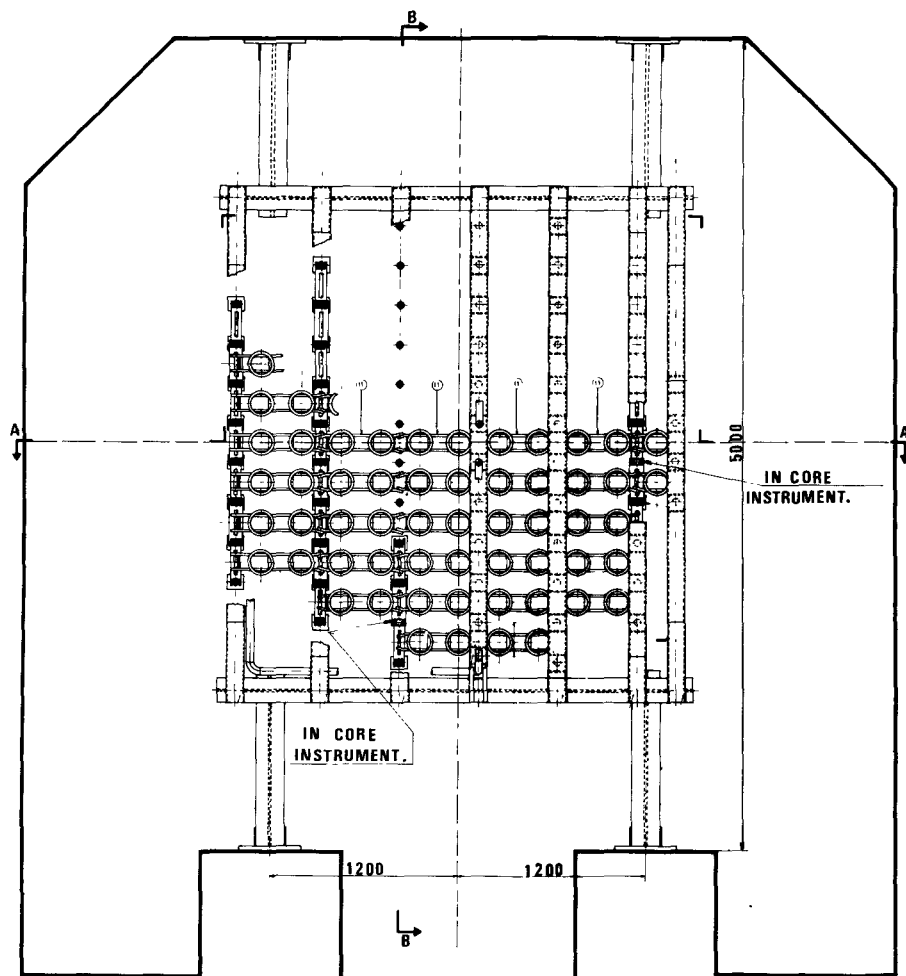


FIG. 2. Garigliano Power Station: support structure of control rod drive housing - plan view.

3. LATINA POWER STATION

On March 17 1969, after some BSD system pipes had been damaged, a television inspection was made of the charge pan and showed that some of the gas thermocouple clips had been dislodged and were missing. An extensive inspection program was immediately started on the reactor high-temperature areas and the steam generators. At the end of the inspection program, there was an overall picture of the abnormal conditions, which enabled a repair program to be worked out.

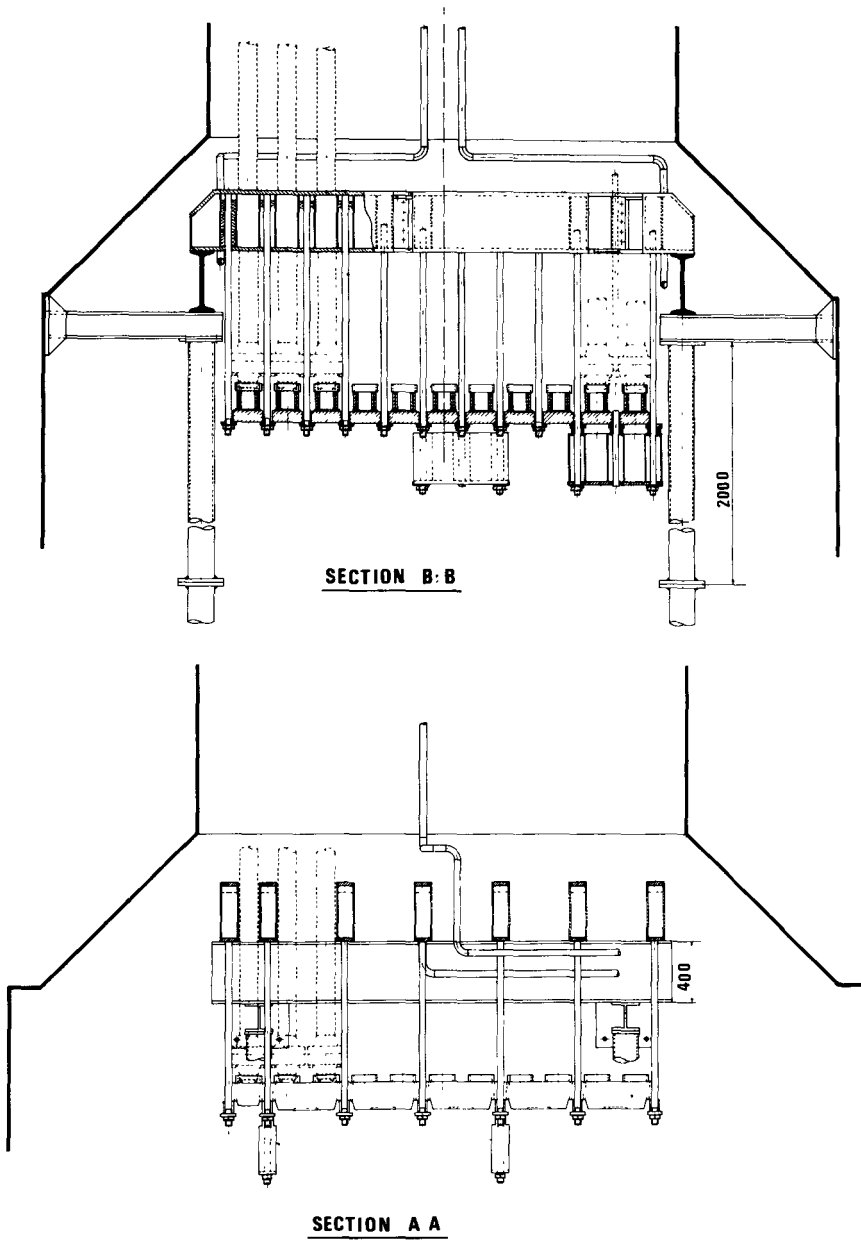


FIG. 3. Garigliano Power Station: support structure of control rod drive housing — side view.

The inspection showed that the ruptures were limited to the reactor zones where the operating temperature was higher than 380°C. The inconveniences which we discovered are to be imputed to an oxidation process to which the carbon steels in CO₂ are particularly subject.

The extent of the oxidation is dependent on the steel composition (silicon content), the fabrication process and the operating temperature. The damaged components were removed. In some cases they were replaced by different systems capable of fulfilling the required function (thermocouple clips and covers) or by components with better characteristics (steam generator bolts and the tie rods).

After the repairs, and on the basis of the safety analysis concerning foreign materials present in the channels and of the assessment of the component conditions after oxidation in a CO₂ high-temperature medium, the reactor was brought into operation again with the following limitations:

(a) reduction of the maximum assessed fuel element temperature (MAFET) from 470 to 450°C;

(b) plant operation with an average gas temperature not higher than 355°C at the reactor outlet; this value should be considered the yearly average, with occasional overshoots conveniently counterbalanced within the limits of (a);

(c) a reduction of the gas outlet temperature, as uniformly as possible over the whole reactor, in particular by reducing the peripheral temperature;

(d) planning inspection of reactor internals at intervals of about one year, to verify the structural conditions and the adequacy of the repairs;

(e) the core containment cage temperature was limited to 360°C; and

(f) installation of a system to scram the reactor in the event of an earthquake.

If, during the next operation, there is an indication of an abnormal situation according to the assumptions of the safety analysis, part or all of the inspections described in (d) will be made immediately.

In addition, it has been decided to continue the collection and examination of information which would be useful for a more precise understanding of the problems and then to revise the above-mentioned operational limitations and provisions on the basis of the policy followed in respect of similar reactors abroad.

4. TRINO VERCELLESE POWER STATION

4.1. General

On April 28 1967, the reactor was shut down for the first refuelling and for general maintenance. Upon opening the vessel, the three sections of the thermal shield were found disconnected and out of place. As a consequence of this failure, an extensive inspection program was performed.

4.2. Results of the inspections

The inspections revealed: failure of about one third of the bolts of the joint connecting the two sections of the lower package; rupture of several

in-core instrumentation guide tubes; rupture of the baskets containing the specimens of vessel steel; rupture of one of the four end boxes of the radial support fastened to the core support plate; erosion of the thermal-shield support brackets and failure of the centering pins of the three shield sections; presence of metal clips in the fuel assemblies.

Subsequent analyses indicated that the thermal shield had become unstable because of the dynamic stresses associated with the primary coolant circulation. This instability had caused the failure of the two core-barrel joint bolts and the damage to the connecting elements of the shield sections.

The failure of the tie-rods of the lower package was caused by various alternating stresses due to the whirlpool created near the tie-rods by the high velocity of the coolant.

4.3. Repairs

The repairs included the following work:

- (a) removal of the thermal shield and of the eight peripheral fuel assemblies;
- (b) replacement of the bolts of the lower package barrel joint with other bolts with better characteristics;
- (c) removal of the tie-rods and, consequently, modification of the connection between the barrel and core-support plate and between this plate and the shroud tubes;
- (d) addition of a secondary core support system.

All the inspection and repair work on the reactor internals was carried out in low radiation areas and with adequate means to minimize personnel exposure.

The achievement of this aim was facilitated by utilizing underwater television cameras and optical equipment such as the borescope and periscope and by performing the repairs with tools specially devised for remote operation. The employment of frogmen both for repairs and for cleaning of the pool facilitated the work considerably.

The surface and air contamination levels in the working areas were monitored and maintained within negligible values. The personnel exposure was therefore well below the limits specified for professionally exposed workers.

4.3.1. Removal of the thermal shield

After removal of the eight fuel assemblies, the neutron flux incident on the vessel is lower than the design value and the thermal stresses due to gamma ray heating are within the levels set forth in the ASME Standards, Section III. Moreover, the removal of the thermal shield entails an appreciable reduction of the forces generated by the primary coolant flow on the reactor internal structures.

4.3.2. New specimen baskets

All the specimens were removed from the reactor and, because of the removal of the thermal shield, they could not be replaced in their

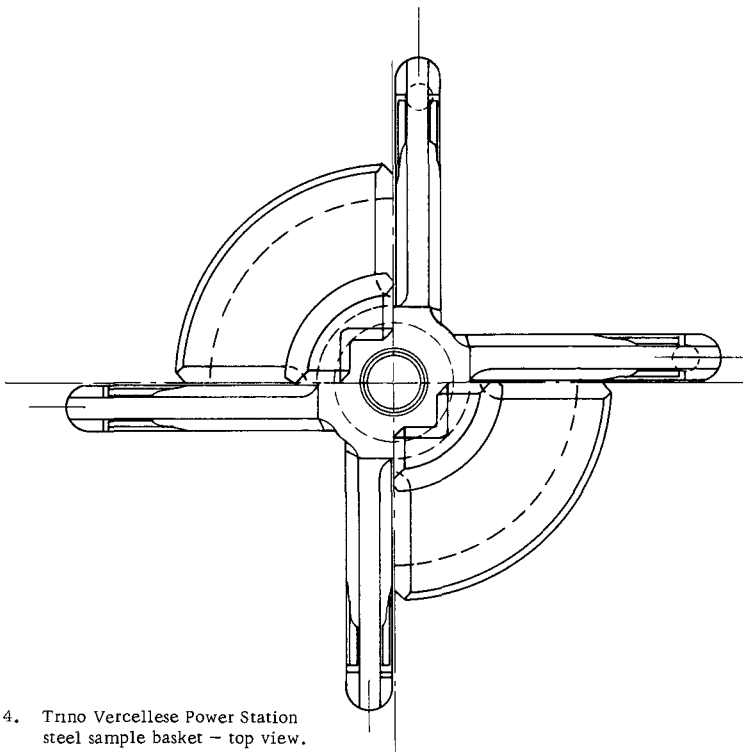


FIG.4. Trino Vercellese Power Station
steel sample basket - top view.

former position. Therefore, the surveillance program was modified and new specimen baskets were placed underneath the core plate in some of the spare control-rod shroud tubes (Figs 4 and 5).

4.3.3. Secondary core support

In addition to repairing the internals, we installed a secondary core support under the lower casting to limit the free fall of the core to less than 20 mm should one of the joints of the lower package fail. This ensures control-rod insertion, reactor scram and the reduction of coolant flow bypass external to the fuel assemblies.

4.4. Tests

4.4.1. General

Before normal plant operation was resumed, special tests were performed to: (a) eliminate any uncertainties about the adequacy of the repairs; (b) verify the adequacy of the calculation methods; (c) obtain information on the behaviour of the structures repaired during plant operation.

The tests were divided into pre-operational tests, operational tests and routine tests.

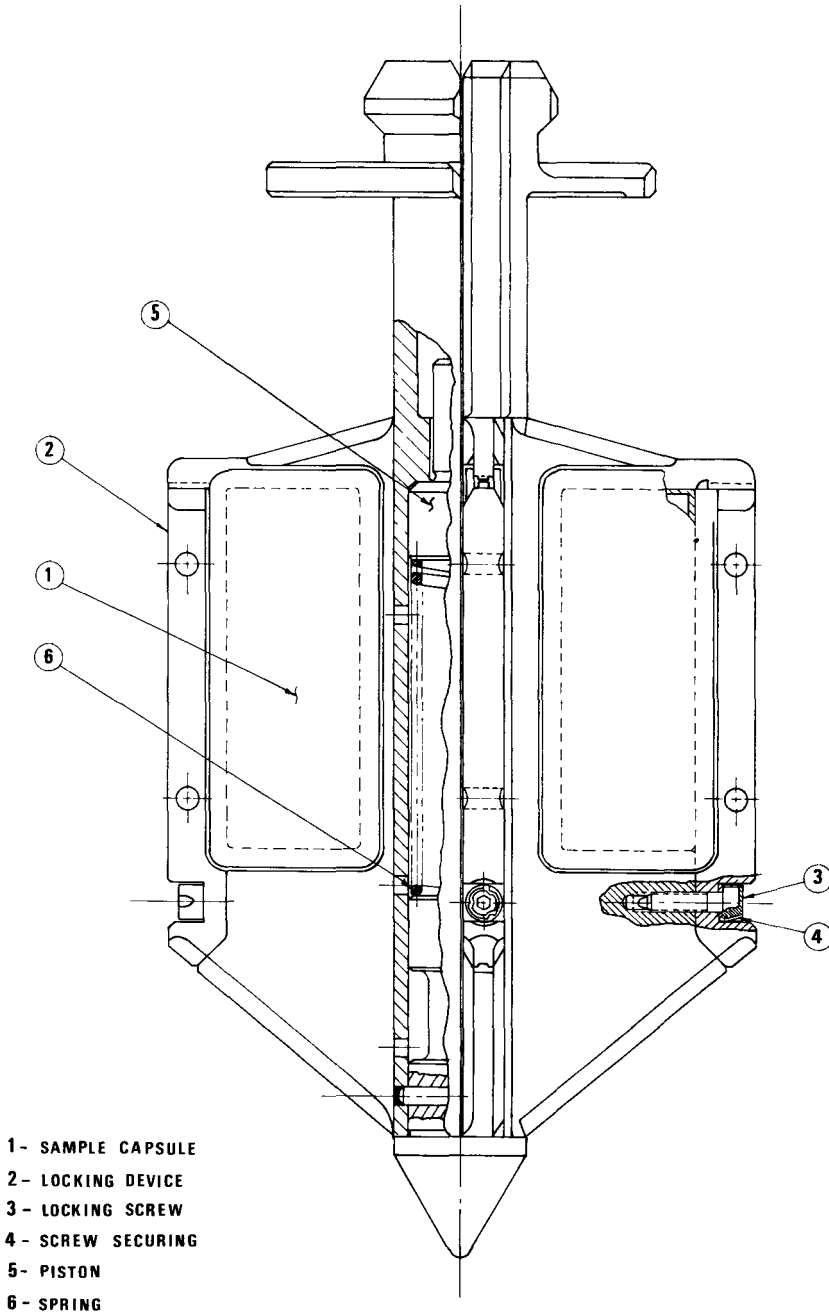


FIG. 5. Trino Vercellese Power Station: steel sample basket - side view.

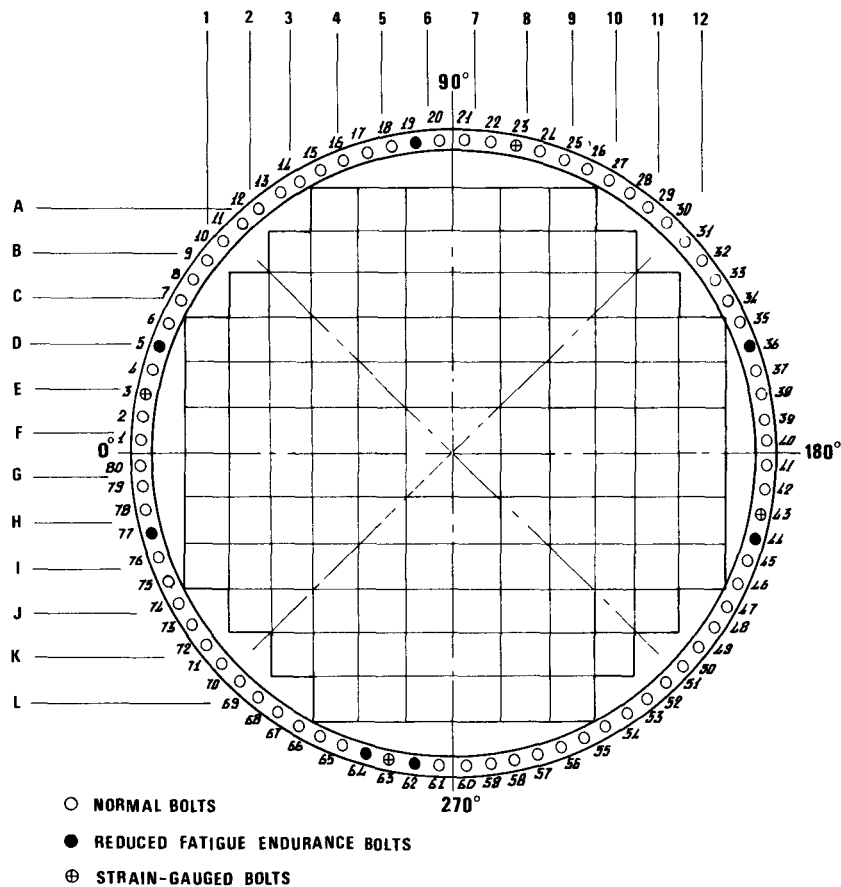
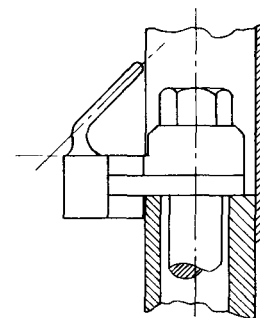
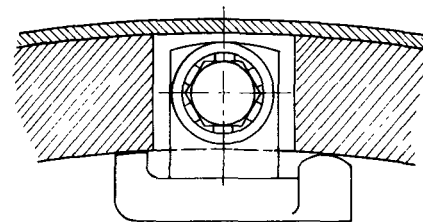


FIG.6. Trino Vercellese Power Station core barrel joint bolts.



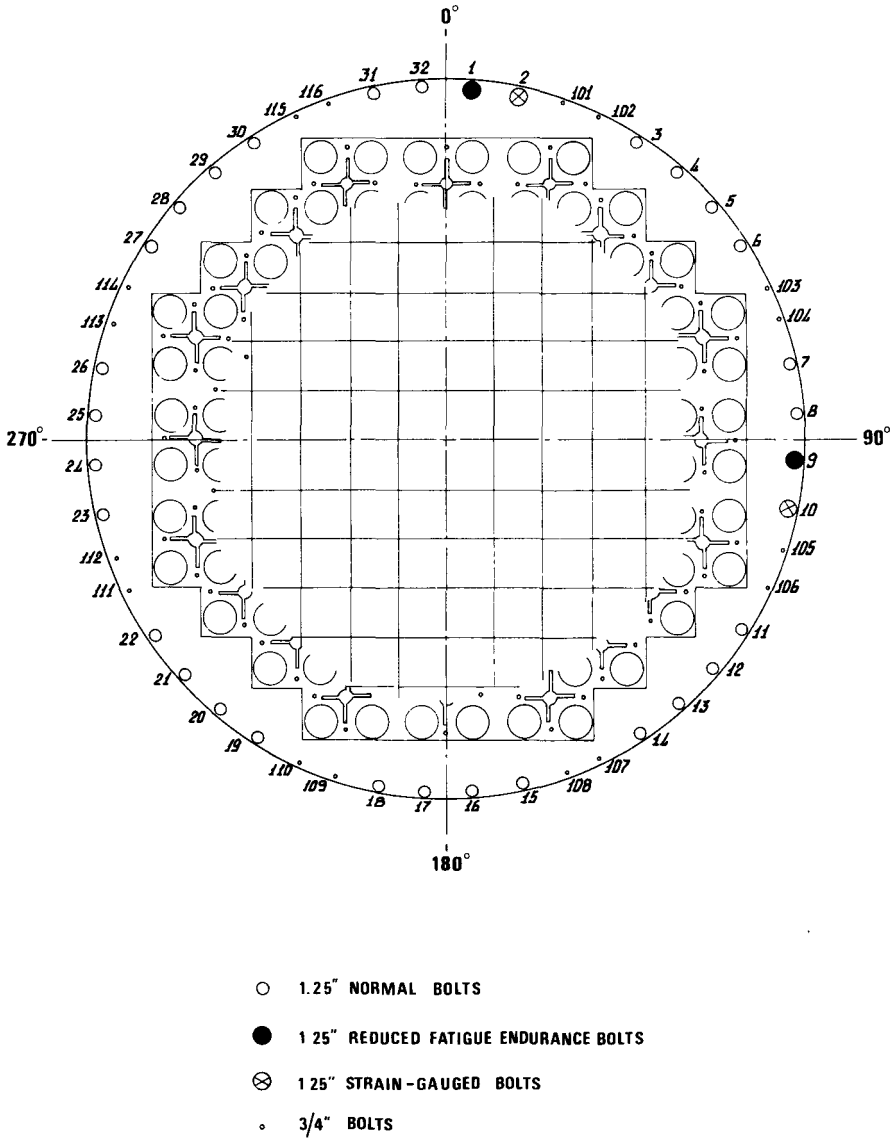


FIG. 7. Trino Vercellese Power Station barrel-plate joint bolts.

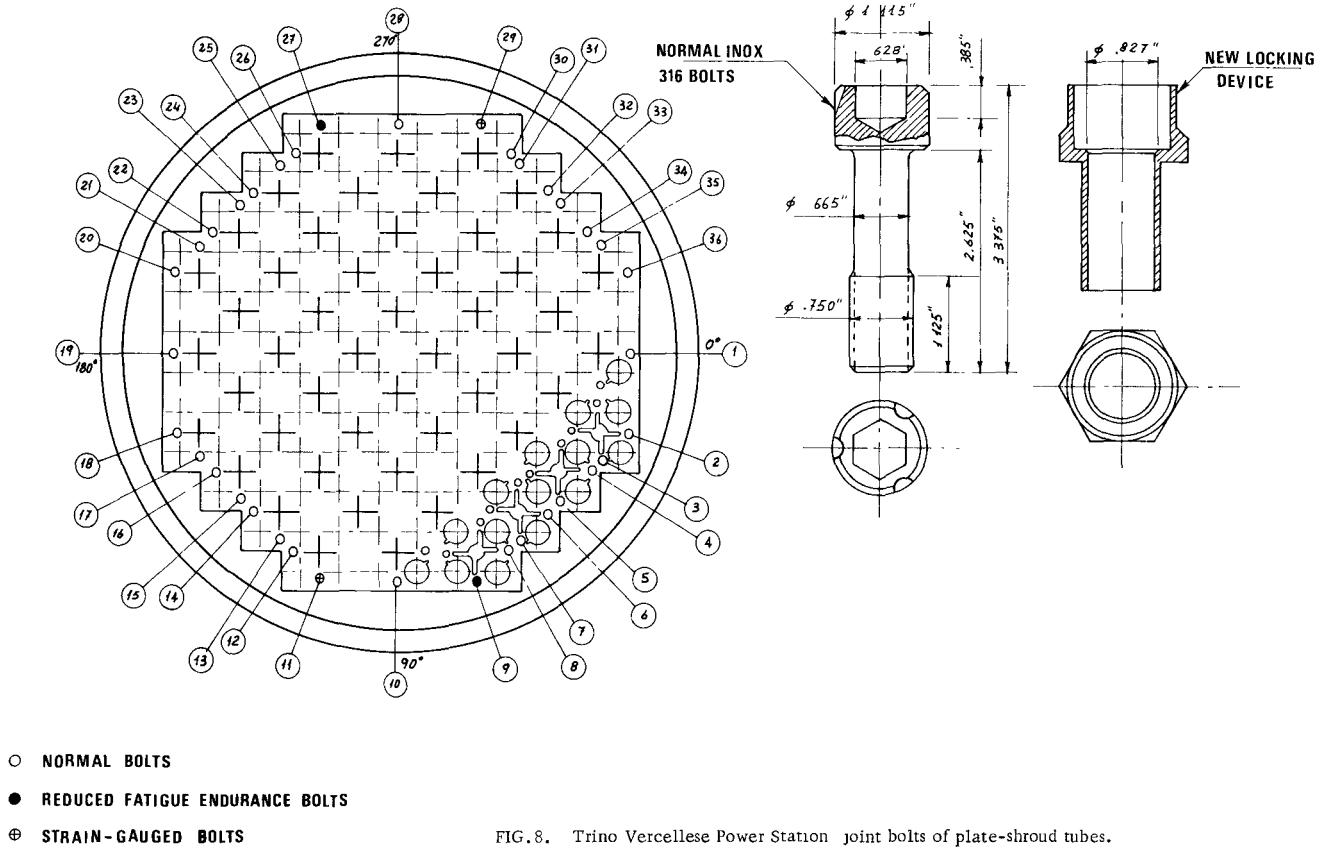


FIG. 8. Trino Vercellese Power Station joint bolts of plate-shroud tubes.

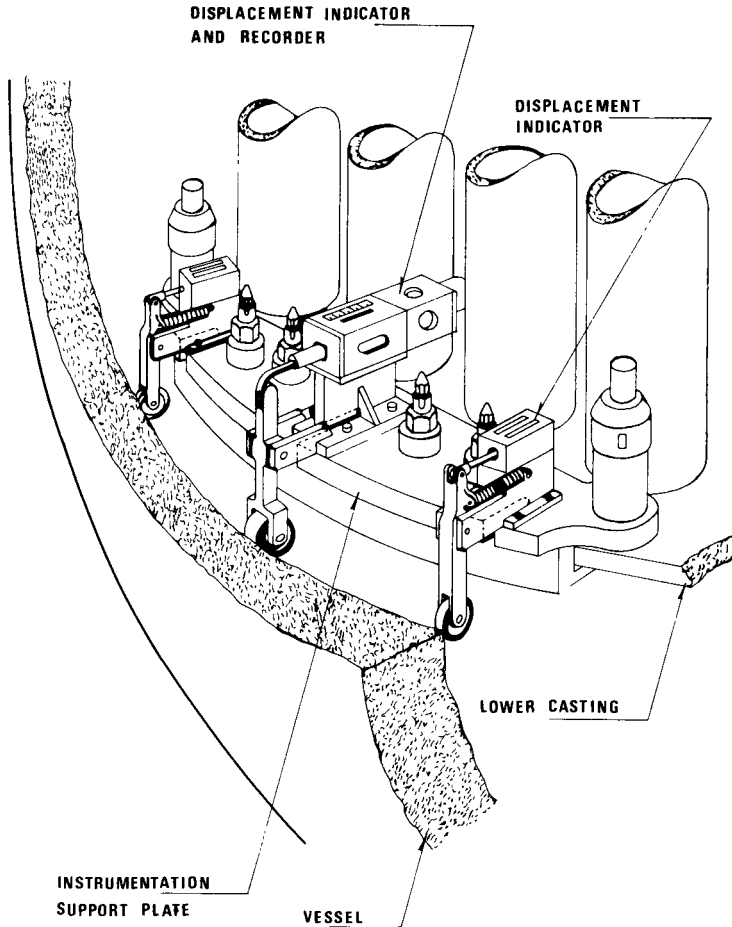


FIG. 9. Trino Vercellese Power Station displacement instrumentation.

4.4.2. Pre-operational tests

All the controls performed before the closure of the vessel are included in this phase, namely: determination of the stresses on the joint bolts and verification of the preloads; measurements of the bending of the core support plate; measurement of clearance between the lower casting and the pressure-vessel wall in the cold condition.

The instrumentation required for these tests comprised strain-gauged bolts, mechanical devices to measure the cold clearance between the lower casting and the pressure-vessel wall, and thermocouples and transducers to measure plate deflection (Figs 6, 7, 8 and 9).

4.4.3. Operational tests

The operational tests included all the checks on the closed pressure vessel, under both cold and hot conditions, with and without the circulation pumps in operation, namely:

- control of the joint behaviour under dynamic load conditions;
- evaluation of the vibration amplitude of the core support plate and frequency residual preload on the bolts;
- evaluation of the clearance between the lower casting and the pressure-vessel wall in the hot condition.

The instrumentation utilized in this phase consisted of:

- notched bolts with reduced fatigue endurance limits (Figs 6, 7, 8);
- mechanical devices to detect any lateral oscillation of the lower package;
- mechanical devices to measure the clearance between the lower casting and the pressure-vessel wall in the hot condition (Fig. 9);
- piezoelectric accelerometers;
- ionization chambers;
- equipment to analyse the nuclear noise and the accelerometer signals.

4.5. Test results

4.5.1. Pre-operational tests

The following conclusions can be drawn from the results:

(a) Deflection of the core support plate

The total deflection of the core support plate was less than expected (0.028 in. instead of 0.055 in.).

(b) Bolts of the barrel joint

The increase in total stress due to core loading was less than expected. The results of the core loading tests confirmed that the applied load has very little effect on the total bolt stresses and therefore the alternating components following a load fluctuation on the core support plates will also be small.

(c) Bolts on core support plate-to-barrel joint

The stress variations due to core loading are well below the estimated values. This result can be attributed to a smaller than expected deflection of the core plate and to better working conditions for the joint than for the model.

(d) Shroud tube bolts

The stress variations in the bolts of the shroud tubes during core loading remained well within the permissible limits, even though they exceeded the estimated values.

4.5.2. Operational tests

The values of the dynamic stresses calculated on the basis of the operational test results were compared with the corresponding estimated values and found to be in good agreement.

(a) Reduced fatigue endurance bolts

At the end of the operational tests, the reduced fatigue endurance bolts and the strain-gauged bolts were removed and inspected. Both types of bolt had maintained the preload applied during assembly so that it was possible to obtain reliable data from the tests.

The inspections showed that the load applied to the reduced fatigue endurance bolts is lower than their static and dynamic endurance. Therefore, the normal bolts will not be damaged during their life if subjected to the design load.

(b) Noise measurements

The accelerometer signals recorded during the various phases of the tests and with a different number of pumps in operation were plotted on frequency spectra. The frequencies of the lines were a multiple of the fundamental frequency of the primary pumps (24.6 Hz). From the examination of all the records, it was concluded that vertical movements of the core support plate do not occur in the frequency range between 5 and 40 Hz. Therefore, the maximum amplitude of the probable vertical displacements of the plate can be assumed to be equal to the minimum value detected by the measurement method, that is ± 0.001 in. (~ 0.025 mm).

The tests confirmed the usefulness of noise analysis techniques in following the behaviour of a reactor, and this means of detection should be developed further.

During the plant life the operational tests will be periodically repeated to obtain useful reference data.

4.5.3. Conclusions

The adequacy of the modifications and repairs performed on the Trino plant was mainly demonstrated by the results of the strain-gauge measurements taken during the pre-operational tests and by the performance of the reduced fatigue endurance bolts.

An extensive program of inspections inside the pressure vessel is now scheduled for the next refuelling shutdown.

4.6. Surveillance program for internals

During the first refuelling shutdown after the repairs, a visual and complete inspection of the internals will be made to ensure that no damage has occurred during operation. In addition to the visual inspection, several bolts from the various joints will be removed to determine the residual preload by the same procedures adopted for the removal of the reduced fatigue endurance bolts and the strain-gauge bolts.

5. CONCLUSIONS

We have been obliged to be very brief in presenting the events of interest in our nuclear plants and thus did not have the opportunity to emphasize all that was relevant. However, when summing up our experience, we believe it necessary to stress the following points:

(1) Extensive knowledge of the plant enables the operating staff to distinguish from normal operation phenomena those which may indicate that an abnormal situation exists.

It is important for the safety engineers of both the licensing bodies and the operators to know the characteristics of the station well enough to evaluate the safety implications of abnormal conditions. This must be joint work, since a decision to resume operation must often be made in a very short time to avoid an undue outage.

(2) It is essential for the plant to be designed so as to allow periodic inspections; most of the inconveniences previously reported were discovered during such inspections and could have led to more severe consequences if the inspections had not been performed.

It is very important for the operating staff or other specialists to be familiar with methods of remote inspection, such as by borescope, television, remotely controlled ultrasonic equipment, and plaster replica methods.

We also realized that it might be useful, during the plant operation, to check the performance of the various systems by means of noise analysis techniques.

(3) It is important for the pre-operational test program to be extensive, since during its implementation abnormal behaviour may be noticed; in any case, a set of the main parameters involved in the safety of the plant should be obtained. These parameters will be compared with those measured during operation after an abnormal condition to understand better the characteristics of the phenomenon or the behaviour of the structures.

(4) When we speak about changes in the design during plant operation, we refer not only to the elimination of defects and abnormal performance, but sometimes also to the necessity of fitting new ideas and concepts into the old plant. This is another reason for keeping the safety engineers of the operating and licensing bodies informed.

(5) The regulatory agencies and operating staffs have to work in close co-operation because, in our opinion, safety problems must be solved clearly beyond any possibility of misunderstandings.

LA SURETE DANS LES CENTRALES DE DOEL ET DE TIHANGE

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Abstract—Résumé—Аннотация—Resumen

SAFETY IN THE DOEL AND TIHANGE NUCLEAR POWER PLANTS.

The Doel (2X390 MW(e)) and Tihange (870 MW(e)) nuclear power plants are being built in densely populated areas, as many nuclear power plants will be in industrialized countries. It is therefore of general interest to examine the studies undertaken in Belgium and the solutions adopted in the choice of equipment and protection systems. At Doel, the twin-unit lay-out has a considerable influence on the choice of safety circuits and devices. Reactor containments have double envelopes: a pressure-resistant steel sphere and a concrete shield cylinder. The safety measures peculiar to the plant (emergency power and coolant water supply, etc.) are treated in detail. The concrete structures inside the containment were designed to conform to the latest safety criteria to avoid aggravating the consequences of a pipe rupture. At Tihange, since the plant is located in a relatively narrow valley, detailed meteorological investigations were undertaken. The authors show how the systematic use of measurements carried out for several years by means of a high mast, and confirmed by traced releases, has made it possible to define the atmospheric dilution coefficient applicable to the plant effluents. The solution adopted for the reactor containment and its associated systems (cooling pool, fuel transfer, spray system, filtration, chimney, etc.) is described. The advantages of a double containment with an inner envelope of pre-stressed concrete and an outer one of reinforced concrete are presented.

LA SURETE DANS LES CENTRALES DE DOEL ET DE TIHANGE.

Les centrales nucléaires de Doel (2X390 MW(e)) et de Tihange (870 MW(e)) sont en construction dans des régions à densité de population élevée, ce qui sera le cas pour l'installation de nombreuses centrales nucléaires dans les pays industrialisés. L'examen des études qui ont été entreprises en Belgique et des solutions qui ont été adoptées dans le choix des équipements et systèmes de protection présente donc un intérêt général. A Doel, la réalisation en unités jumelles a des répercussions importantes sur les circuits et dispositifs de sauvegarde. Les enceintes étanches sont à double enveloppe: une sphère métallique résistant à la pression et un cylindre en béton de blindage. Les sécurités propres de la centrale (alimentation de secours en électricité et eau de refroidissement, etc.) sont décrites en détail. Les structures en béton à l'intérieur de l'enceinte ont été conçues pour tenir compte des derniers critères de sécurité pour éviter l'aggravation des conséquences d'une rupture de tuyauterie. A Tihange, la centrale étant située dans une vallée encaissée, des études météorologiques détaillées ont été entreprises. Il est montré comment l'exploitation systématique des mesures effectuées depuis plusieurs années à l'aide d'un haut mât, confirmée par une campagne de rejets tracés, a permis de définir le coefficient de dilution atmosphérique applicable aux rejets de la centrale. La solution retenue pour l'enceinte de confinement du réacteur et ses systèmes associés (piscine de désactivation, transfert du combustible, aspersion, filtration, cheminée, etc.) est décrite. Les avantages présentés par une double enceinte, dont l'enveloppe intérieure est en béton précontraint et l'enveloppe extérieure en béton armé, sont exposés.

ТЕХНИКА БЕЗОПАСНОСТИ НА АТОМНЫХ ЭЛЕКТРОСТАНЦИЯХ ДООЛЬ И ТИАНЖ.

Атомные электростанции Доэль (2 × 390 МВт (эл)) и Тианж (870 МВт (эл)) строятся в настоящее время в густонаселенных районах, как и многие другие атомные электростанции, которые будут построены в промышленно развитых странах. Поэтому представляет интерес рассмотреть результаты проведенных в Бельгии исследований и принятые решения по выбору оборудования систем защиты. Используемая на атомной электростанции в Доэле компоновка с двумя блоками оказала значительное влияние на систему защиты и оборудование. Реакторы имеют двойную защитную оболочку: стальная сферическая обо-

лочка и бетонный защитный цилиндр. Подробно рассматриваются условия обеспечения безопасности атомных электростанций (аварийный источник питания, аварийное снабжение охлаждающей водой и т. д.). Бетонные конструкции внутри защитной оболочки были спроектированы с учетом современных критериев безопасности с целью предотвращения серьезных последствий в случае разрыва трубопроводов. Проведены детальные метеорологические исследования для атомной электростанции Тяньцзинь, поскольку станция расположена в относительно узкой долине. Показано, как систематическое использование результатов измерений, проводившихся в течение нескольких лет на высокой мачте и подтвержденных экспериментами с выбросом в атмосферу меченых атомов, позволило определить коэффициент атмосферного разбавления применительно к выбросам атомной электростанции. Описывается решение, принятое по защитной оболочке реактора и связанным с ней системам (бассейн выдержки, система перегрузки топлива, система разбрызгивания, фильтры, вытяжная труба и т. д.). Показаны преимущества использования двойных защитных оболочек (с внутренней оболочкой из предварительно напряженного бетона и с внешней оболочкой из железобетона).

LA SEGURIDAD EN LAS CENTRALES NUCLEARES DE DOEL Y TIHANGE.

Las centrales nucleares de Doel (2x390 MW(e)) y Tihange (870 MW(e)) se están construyendo en zonas densamente pobladas, como ocurrirá con numerosas centrales nucleares que se construyan en países industrializados. La consideración de los estudios realizados en Bélgica y de las soluciones adoptadas para la selección de equipos y sistemas de protección es, pues, de un interés general. En Doel la disposición en unidades gemelas tiene repercusiones importantes sobre los circuitos y dispositivos de seguridad. Los sistemas de contención estancos son de doble capa: una esfera metálica para resistir la presión y un cilindro de hormigón para blindaje. Se detallan los sistemas de seguridad propios de la central (alimentación de emergencia de energía eléctrica y agua de refrigeración, etc.). Las estructuras de hormigón en el interior del recinto de contención se han diseñado teniendo en cuenta los últimos criterios de seguridad para evitar un agravamiento de las consecuencias de una ruptura del sistema de tubos. En Tihange se han llevado a cabo estudios meteorológicos detallados por estar situada la central en un valle cerrado. Se explica cómo la utilización sistemática de las medidas efectuadas desde hace varios años con ayuda de un mástil elevado y su comprobación en una campaña de emisiones marcadas, ha permitido definir el coeficiente de dilución atmosférica, aplicable a los efluentes gaseosos de la central. Se describe la solución adoptada para la contención del reactor y los sistemas asociados (piscina de desactivación, transferencia de combustible, aspersión, filtración, chimenea, etc.). Se describen las ventajas que presenta un sistema doble de contención, con cubierta interior de hormigón pretensado y cubierta exterior de hormigón armado.

1. GENERALITES

Peu après la première conférence de Genève, en 1955, une petite centrale expérimentale du type à eau sous pression (PWR) avait été commandée par l'industrie belge à la société américaine Westinghouse. A l'origine, cette centrale devait être installée dans les abords immédiats de Bruxelles. Le choix d'un site urbain déclencha certaines oppositions et les travaux d'aménagement furent suspendus, pour être repris en 1957, après qu'un nouveau site eut été choisi à Mol, sur les terrains du Centre d'étude de l'énergie nucléaire.

Cette petite centrale de 10,5 MW(e), dont le démarrage fut retardé par le délai de ratification de la loi nationale d'assurances du risque nucléaire, fut mise en service en 1962 et se révéla très utile pour la formation du personnel d'exploitation des centrales ultérieures. La petite taille de l'unité la rendait particulièrement apte à jouer ce rôle.

Pour leur première réalisation de taille industrielle, les producteurs belges d'électricité décidaient en 1961 de construire en commun avec Electricité de France une centrale PWR de 266 MW(e) installée à Chooz près de la frontière franco-belge. Démarrée en 1967, cette centrale fut

handicapée par un accident aux pièces internes du réacteur, auquel un mémoire est consacré.¹ Depuis sa remise en service l'exploitation de la centrale des Ardennes est très satisfaisante.

En 1965, les producteurs belges d'électricité, à savoir les sociétés EBES, ESMALUX, INTERBRABANT, INTERCOM et UCE LINALUX-HAINAUT, décidèrent de mettre à l'étude l'installation à Doel et à Tihange de deux centrales nucléaires de grande puissance.

Si la réduction des prix spécifiques incitait à accroître la puissance unitaire des centrales, la constitution du réseau, équipé d'unités classiques de 125 MW(e) en service et d'unités de 300 MW(e) en construction, grevait le prix du kWh produit par une grosse unité nucléaire du coût de la réserve qui devait lui être associée.

C'est la raison pour laquelle la préférence fut donnée, pour la centrale du Doel, à des unités de 390 MW(e). La centrale, propriété indivise des sociétés mentionnées ci-dessus auxquelles s'est jointe l'UKEC, comporte deux chaudières nucléaires à deux boucles, chaque chaudière alimentant un turbo-alternateur de 390 MW(e). Elle est conçue suivant le principe des centrales jumelées («twins»), une série d'auxiliaires étant communs ou partagés entre les deux unités.

A Tihange, la décision d'Electricité de France de participer au projet, en symétrie à sa participation à la centrale franco-belge de Chooz, permet le choix d'une seule chaudière nucléaire à trois boucles, alimentant deux groupes turbo-alternateurs d'une puissance totale de 870 MW(e). Le réseau français absorbera la moitié de la production de la centrale et la conclusion d'un contrat de secours mutuel entre producteurs français et belges procure un avantage supplémentaire à cette réalisation commune.

Les réacteurs de Doel et de Tihange sont des PWR, permettant ainsi une mise à profit optimale des enseignements tirés de l'exploitation de BR3 et de la centrale des Ardennes. Les unités de Doel doivent démarrer respectivement en 1973 et en 1975; la centrale de Tihange sera mise en service en 1974.

Les producteurs belges d'électricité ont confié à l'Association Momentanée Trabel (Traction Electricité - Electrobél) la mission d'ingénieur conseil. Au sein de cette association Traction Electricité est chargée des études de la centrale de Doel tandis qu'Electrobél, associée à EDF, exécute la même mission en ce qui concerne la centrale de Tihange.

En fait de sécurité, beaucoup de principes sont communs aux deux centrales. Ils résultent des sujétions voisines en matière d'environnement auxquelles les deux installations, distantes de 115 km à vol d'oiseau, sont soumises; ils résultent aussi de l'application des consignes de sécurité américaines récentes et de l'expérience acquise en commun dans les centrales de Mol et de Chooz.

La densité de population le long de la côte et des fleuves belges est élevée. De ce fait les sites nucléaires ne présentent pas les caractéristiques d'isolement que l'on peut trouver dans les pays de grandes dimensions géographiques. Les sites de Doel et Tihange sont localisés, le premier le long de l'Escaut à 15 km en aval de l'agglomération anversoise, le deuxième dans la vallée peuplée de la Meuse à 25 km en amont de Liège.

¹ KIEFFER, J., et al., ces Actes, mémoire 575, vol. 2.

Dans les deux centrales, ces conditions ont nécessité de recourir à un confinement plus élaboré du réacteur; elles ont conduit au choix d'une double enceinte d'isolement avec mise sous dépression et filtration de l'espace entre enceintes.

On reviendra dans la suite de l'exposé sur la solution pratique qui a été retenue pour le confinement des réacteurs de Doel et Tihange et sur les études qui y ont conduit.

Les deux centrales ont été conçues en tenant compte de l'expérience américaine, caractérisée notamment par:

- les spécifications de l'USAEC, et plus particulièrement celles du document 10 CFR Part 50, General Design Criteria for Nuclear Power Plant Construction Permit;
- les recommandations de l'ANS: Nuclear Safety Criteria for the Design of Stationary Pressurized-Water Reactor Plants;
- les codes suivants: ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels,
Draft ASME Code for Pumps and Valves for Nuclear Power,
USAS B 31.7 Nuclear Power Piping.

En dernier lieu, il y a lieu de souligner la mise à profit de l'expérience vécue lors de la réparation de la centrale de Chooz dans la conception des piscines. D'une part l'accès au dispositif de transfert du combustible irradié a été facilité, d'autre part on a prévu la possibilité d'accéder au réacteur pour les inspections de la cuve, les pièces internes étant maintenues sous eau dans une partie de la piscine isolée par un batardeau.

2. CENTRALE DE DOEL

2.1. Conception générale

La centrale est bâtie sur la rive gauche de l'Escaut, dans une zone appelée à un important développement portuaire et industriel.

Parmi les éléments qui ont contribué au choix des solutions adoptées il y a lieu de citer plus spécialement:

- la proximité de la ville d'Anvers
- le passage sur le fleuve de navires importants
- l'existence d'une industrie pétrolière aux environs
- le degré de pollution de l'eau du fleuve
- l'existence de marées importantes.

Ces éléments ont en particulier dicté le choix des solutions adoptées en ce qui concerne la protection contre l'accident de référence (double enceinte) et la conception des circuits de réfrigération.

D'autre part, la décision ayant été prise de construire deux unités, il s'est avéré plus intéressant, tant techniquement qu'économiquement - vu le décalage de deux ans - de concevoir les deux unités comme un ensemble et de partager certains auxiliaires (centrales jumelées).

C'est dans cet esprit qu'a été conçu le plan d'ensemble de la centrale de Doel. On voit sur la figure 1 les deux enceintes de réacteur, de part et d'autre d'un bâtiment commun d'auxiliaires nucléaires, perpendiculaire à un ensemble comportant les salles d'auxiliaires électriques et la salle des machines.

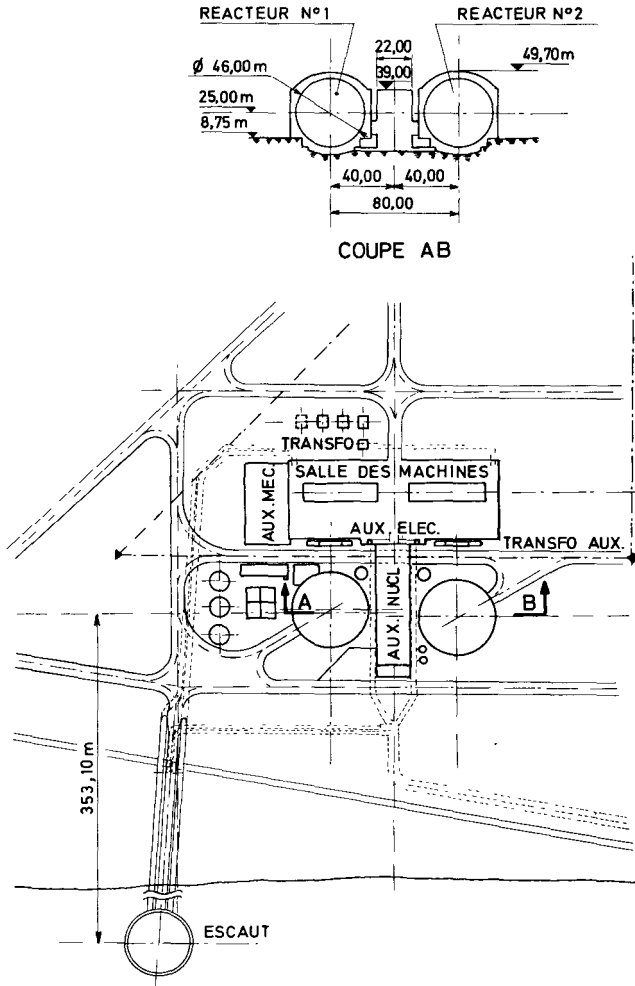


FIG.1. Centrale nucléaire de Doel: disposition générale.

Le bâtiment des auxiliaires nucléaires abrite:

- les auxiliaires propres à chacun des réacteurs: circuit de refroidissement du réacteur à l'arrêt, d'injection de sécurité basse pression et de recirculation, circuit de contrôle volumétrique
- les auxiliaires partagés par les deux réacteurs: refroidissement intermédiaire, injection de sécurité haute pression, aspersion
- les auxiliaires communs: préparation de solution de bore, stockage des effluents liquides et gazeux, piscine de désactivation.

Chacun des bâtiments des chaudières nucléaires abrite un réacteur PWR de conception Westinghouse, fourni par l'association ACEC-Cockerill, à deux boucles comportant chacune un générateur de vapeur de 44 000 pieds carrés. Les ouvrages ont été conçus de manière à permettre l'introduction en un seul élément de ces générateurs et à éviter la soudure sur place des enveloppes.

2.2. Confinement

Le confinement en cas d'accident est basé sur le principe de la double enceinte. L'enceinte intérieure est réalisée sous forme d'une sphère métallique de 46 m de diamètre, conçue pour résister aux conditions résultant de la rupture double de la plus grosse tuyauterie primaire ($2,9 \text{ kgf/cm}^2 \text{ eff.}$, 129°C). Le taux de fuite maximal admis est de 0,1% du volume contenu par jour, dans les conditions de l'accident de référence.

L'enceinte extérieure est constituée d'un cylindre de béton de 1 m d'épaisseur, avec dôme sphérique de 70 cm d'épaisseur. L'ensemble a 50 m de diamètre extérieur et s'élève à 42 m au-dessus du sol. Le taux de fuite maximal admis de cette enceinte est de 1% par jour du volume total enfermé pour une pression différentielle de 350 mm CE. L'étanchéité est réalisée par un revêtement retenu à la suite d'une série d'essais de laboratoire. L'enceinte secondaire est protégée contre une surpression accidentelle par une garde hydraulique de grandes dimensions, localisée à l'emplacement de l'accès du matériel.

Entre l'enceinte intérieure en acier et l'enceinte extérieure en béton est ménagé un espace intermédiaire maintenu en permanence en dépression, soit par des compresseurs, soit par des ventilateurs de brassage.

Les compresseurs aspirent l'air directement dans l'espace intermédiaire et le refoulent

- en situation normale, dans une gaine équipée de filtres vers l'aspiration des ventilateurs d'extraction du bâtiment des auxiliaires nucléaires;
- en situation accidentelle, vers l'enceinte intérieure en acier; cette disposition permet d'éviter tout relâchement d'activité vers l'extérieur.

Il y a deux compresseurs, chacun servant de réserve à l'autre. Ils sont capables de refouler chacun $85 \text{ Nm}^3/\text{h}$ contre une pression de $3 \text{ kgf/cm}^2 \text{ eff.}$

Les ventilateurs permettent d'effectuer un brassage de l'air de l'espace intermédiaire au taux de un renouvellement par heure tout en maintenant cet espace en dépression. Une faible partie de leur débit, correspondant aux entrées d'air par les inétanchéités de l'enceinte extérieure en béton, est dérivée par la gaine de refoulement des compresseurs vers l'aspiration de ventilateurs d'extraction du bâtiment des auxiliaires nucléaires.

Les ventilateurs sont pourvus de filtres à iode; l'idée de base est identique à celle qui a prévalu à Tihange et qui sera examinée au paragraphe 3.2 de ce mémoire.

Les pénétrations de tuyauteries dans l'enceinte primaire sont du type conventionnel pour les enceintes métalliques; elles sont prévues de manière à permettre le contrôle individuel lors d'un arrêt de l'installation.

Les pénétrations électriques sont équipées de carottes en araldite, placées dans des manchettes, et pourvues de deux «0 rings» permettant également un essai individuel.

Les tuyauteries sont, dans le cas de Doel, ancrées à l'enceinte secondaire et traversent librement des manchettes de l'enceinte primaire, auxquelles elles sont raccordées par des joints de dilatation prévus pour compenser les mouvements relatifs des deux enceintes d'une part, et d'autre part les mouvements des tuyauteries depuis leur point d'ancrage.

Une attention particulière a été attachée à la tuyauterie de vapeur: une rupture dans l'enceinte intermédiaire amènerait, malgré la garde hydraulique de l'enceinte secondaire, une montée en pression qui compromettrait la tenue des deux enceintes. Diverses solutions ont été examinées; la solution retenue comporte deux tuyauteries concentriques, frettées à chaud. Une rainure hélicoïdale permet de détecter une fuite du tube central. La présence du tube extérieur empêche cependant le déboîtement de la tuyauterie interne, ce qui garantit la tenue de l'enveloppe extérieure.

Les dispositifs habituels de sauvegarde sont prévus: aspersion et ventilation. Les circuits sont conçus avec les redondances nécessaires pour garantir leur efficacité en cas de défaillances d'équipement. Différentes précautions sont prises cependant pour éviter une aspersion accidentelle (redondance de circuits à coïncidence, émission de courant, manœuvres simultanées de pompe et vanne, signal de pression d'enceinte).

Les structures en béton à l'intérieur de l'enceinte étanche ont été déterminées en respectant les règles générales dont il est fait état au paragraphe 3.2 ci-après. Il y a cependant lieu de signaler que la complexité des structures internes a amené le bureau d'études à entamer, avec la collaboration du laboratoire de résistance des matériaux de l'Université de Bruxelles, une série d'essais sur modèle réduit (essais à l'aide de jauges de contrainte, essais photo-élastiques et construction d'un modèle en mousse). En particulier, la dalle support des générateurs de vapeur est une plaque épaisse déconnectée. Des études par jauges ont permis de déterminer l'état des contraintes sur les faces inférieures et supérieures; la position de l'axe neutre et la variation des contraintes dans l'épaisseur ont été déterminées par des essais photo-élastiques. Des expériences extensométriques ont été effectuées ensuite sur certaines parties critiques, telles que les percements de petit diamètre pour le passage des tuyauteries. Les piliers supports de cette dalle sont inclinés et à inertie variable.

Des jauges de contrainte ont été mises en place pour suivre in situ l'état de tension des armatures, notamment de la dalle inférieure, pour vérifier le bien-fondé des méthodes utilisées.

2.3. Dispositions particulières

Comme dans toutes les centrales nucléaires, des groupes diesels permettent, en cas de défaillance des sources normales

- en situation normale, d'amener les centrales à l'arrêt à froid
- en situation accidentelle, d'alimenter le nombre requis de dispositifs de sauvegarde.

D'une manière générale, ces dispositifs sont conçus sur la base d'une symétrie par 4, 2 des 4 engins ayant une capacité suffisante pour assurer la fonction.

De même, afin de respecter cette symétrie, l'alimentation de secours comporte 4 groupes diesels, qui peuvent être connectés aux 4 jeux de barre vitaux, alimentés en condition normale par moitié par l'un et l'autre groupe.

D'autre part, une attention particulière a été apportée à la continuité d'alimentation en eau de refroidissement. En premier lieu, pour éviter l'envasement et les corps flottants, la prise d'eau est située à mi-hauteur entre le lit et le niveau des plus basses eaux, dans une tour située dans la rivière. Cette tour est protégée contre le choc de navires par des ducs d'Albe. Cependant la solution ne paraissait pas entièrement sûre; en particulier un incendie d'une nappe d'hydrocarbures pourrait compromettre la continuité d'alimentation. Pour faire face à toutes les éventualités, il a été décidé d'associer une tour de réfrigération à chacun des quatre échangeurs de refroidissement intermédiaire, en schéma monobloc. Le circuit de refroidissement intermédiaire comprend deux collecteurs, alimentant chacun la moitié des dispositifs de sauvegarde de chacune des centrales. L'importance attachée à ce circuit s'explique d'autant mieux si l'on tient compte du fait que la pollution possible de l'eau a amené à alimenter par ce circuit tous les équipements à refroidir.

Enfin, la centrale a été protégée contre les dégâts dus à une inondation venant du fleuve par un remblai à un mètre au-dessus du niveau des plus hautes eaux (raz de marée de 1953). En outre des précautions ont été prises pour éviter la paralysie des circuits de sauvegarde par une inondation résultant d'une rupture de tuyauterie. Dans cet esprit, les boîtes à eau des échangeurs de réfrigération intermédiaire sont situées à l'extérieur du bâtiment des auxiliaires; une rupture des tuyauteries d'eau brute ne risque pas d'inonder le bâtiment. De plus, la cave contenant les pompes de refroidissement réacteur à l'arrêt, utilisées également en phase de recirculation, est équipée de seuils aux différents passages et de puisards dans les différentes salles, pourvus d'un dispositif séquentiel de détection d'alarmes. La présence des seuils permet d'étaler sur la surface totale de la cave le volume d'eau perdu, tout en permettant la détection à distance de l'origine de la fuite. Le local est dans un tel cas isolé pour éviter la dispersion de produits radioactifs; la ligne de recirculation utilisée est mise hors service par action automatique, l'opérateur pouvant rétablir un circuit en fonction de la localisation de la fuite. Des moteurs refroidis à l'eau permettent la reprise du service dans un local à atmosphère confinée.

2.4. Inspection en service

La centrale est conçue de manière à permettre l'inspection en service, en principe suivant les dispositions de la section XI du code de l'ASME (Boiler and Pressure Vessel Code), et en utilisant les possibilités offertes par ce code. En particulier, afin de réduire la formation d'argon-41, l'espace entre béton et cuve au niveau du cœur est réduit, et de ce fait, l'inspection de la cuve devra être réalisée depuis la face interne.

Au-dessus du niveau du cœur, une accessibilité externe maximale est ménagée de manière à permettre le libre accès à toutes les tubulures de la cuve et aux supports de celle-ci, reposant sur un anneau scellé dans le béton; un matériau absorbant posé sur cet anneau diminue les doses d'irradiation dans cette zone.

3. CENTRALE DE TIHANGE

3.1. Conception générale

La centrale est bâtie en aval de Huy, sur la rive droite de la Meuse. Il a été fait appel pour la chaudière nucléaire à la plus récente technique

Westinghouse, qui n'est appliquée, à ce niveau de puissance, que dans des chaudières dont la construction se poursuit parallèlement aux Etats-Unis et à Fessenheim en France; fournie par l'association ACECO (ACEC-Cockerill)-SFAC-FRAMATOME, elle développe par trois boucles de refroidissement une puissance thermique de 2660 MW.

Les circuits de refroidissement des condenseurs des deux turbines sont équipés de réfrigérants atmosphériques exploitables en circuit ouvert ou fermé; le respect de la limite de température imposée en aval de la centrale nécessite leur fonctionnement 1 à 2 mois par an.

3.2. Confinement

3.2.1. Etudes météorologiques

Dans la région de Tihange, la vallée de la Meuse est orientée dans le sens des vents dominants. La population est assez élevée: quelque 500 personnes résident à moins de 1 km du réacteur, et des agglomérations de 1000 à 10 000 habitants en sont distantes de moins de 3 km. Cette vallée est bordée de collines dont les sommets s'élèvent de 100 à 200 m au-dessus du niveau du site de la centrale; certaines d'entre elles, relativement peuplées, sont à moins de 2 km du réacteur.

Dès le début des études ces éléments ont mis en évidence la nécessité de recourir à un confinement poussé du réacteur de nature à éviter tout risque d'évacuation de population en cas d'accident majeur. Une double enceinte intégrale a été retenue, cette conception étant la seule qui permette de canaliser sous dépression toutes les fuites potentielles de l'enceinte intérieure et de les rejeter après filtration des halogènes.

La détermination finale des caractéristiques des divers équipements intervenant dans le confinement dépendait notamment de la connaissance des conditions de dilution atmosphérique; par ailleurs le ministère belge de la Santé publique avait demandé qu'il soit procédé à des mesures météorologiques directes sur le site de Tihange. Ce programme fut conduit en collaboration avec le Centre d'étude de l'énergie nucléaire de Mol à l'aide d'un mât météorologique de 200 m de hauteur, monté sur le site et exploité de 1967 à 1970. Le mât est équipé de six anémomètres, de six thermocouples et trois girouettes échelonnées entre les niveaux 8 et 200 m.

L'emploi à Tihange du modèle mathématique gaussien du Centre d'étude de l'énergie nucléaire [1] entraînait deux incertitudes. Les écarts-types des distributions transversales sont des fonctions exponentielles de la distance dont les coefficients, liés à un paramètre de stabilité incluant vitesse du vent et gradient thermique vertical, ont été déterminés à la suite d'une étude originale effectuée sur le site plan de Mol [2]; l'applicabilité des résultats de cette étude au site accidenté de Tihange n'était pas évidente a priori. D'autre part le modèle ne tient pas compte du relief; le calcul de la dilution en un lieu déterminé est fait en considérant la hauteur relative du rejet par rapport à ce lieu, ce qui suppose implicitement que la trajectoire du plumet n'est pas influencée par le relief, hypothèse a priori très conservatrice. Une campagne de rejets tracés effectuée à la hauteur de 150 m a permis de s'assurer que le modèle mathématique pouvait néanmoins être utilisé avec une approximation très satisfaisante. En effet le rapport des coefficients de dilution $\delta^{-1} \text{ s/m}^3$

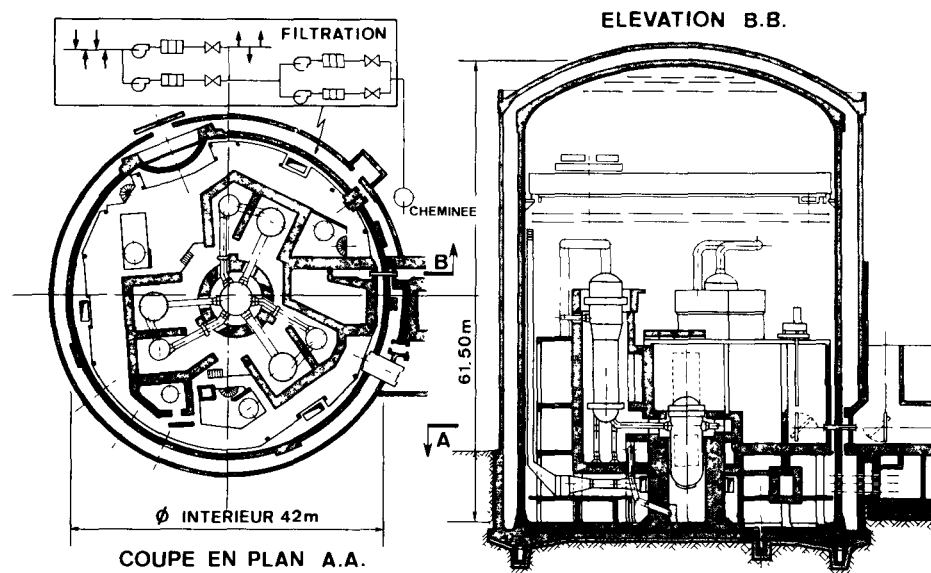


FIG.2. Centrale nucléaire de Tihange: bâtiment du réacteur.

obtenus d'une part par le calcul théorique, d'autre part par les rejets tracés, est compris le plus souvent entre 1 et 5 pour les points de la vallée et entre 1 et 22 pour ceux du relief, avec une majorité de résultats inférieurs à 7,5; il faut noter que la méthode utilisée pour les rejets tracés peut mener à sous-estimer les concentrations réelles d'un facteur 2 environ.

Le dépouillement des observations des trois années porte sur 16 points choisis pour leur proximité du réacteur, leur altitude ou la densité de population avoisinante.

Pour chacun de ces points, les facteurs de dilution et de dépôt instantanés et moyens les plus défavorables sont déterminés pour des hauteurs de cheminée 100 et 160 m. Le modèle mathématique utilisé permet d'obtenir les facteurs applicables à divers intervalles d'intégration compris entre 3 h et 1 an; ces intervalles sont déplacés mois par mois sur les 3 ans pour la recherche de l'année la plus défavorable, et, sur cette dernière année, pour la recherche du mois le plus défavorable. Pour les intervalles d'intégration inférieurs au mois, la recherche est faite dans le mois le plus défavorable. Une étude statistique effectuée pour les points les plus critiques et portant sur les trois années de mesure a permis de vérifier que les périodes les plus défavorables de 1 jour et 10 jours déterminantes pour l'étude de l'accident étaient bien comprises avec des probabilités respectives de 99,48% et 100% à l'intérieur du mois le plus défavorable; ce résultat pouvait être pressenti pour des raisons évidentes de corrélation mais devait néanmoins être vérifié.

Le dépouillement met en évidence la canalisation des vents par la vallée; les heures d'occurrence de vents de directions perpendiculaires à la vallée sont moins nombreuses à 80 m de hauteur qu'à 200 m, près de deux fois moins pour certains secteurs. A 200 m de hauteur, la comparaison avec les résultats obtenus à 120 m du sol à Mol montre que l'effet de vallée n'est plus sensible.

D'autre part la configuration de la vallée rend préoccupants les phénomènes de stagnation d'effluents qui pourraient s'y produire.

L'état d'avancement des études ne permet pas encore de fixer la hauteur de la cheminée, qui sera cependant supérieure à celle des crêtes avoisinantes.

3.2.2. Structures et équipements

La double enceinte du bâtiment du réacteur est réalisée par une enveloppe intérieure en béton précontraint de 42 m de diamètre interne et de 70 cm d'épaisseur, et une enveloppe extérieure en béton armé de 40 cm d'épaisseur. La largeur de l'espace entre enveloppes est de 2 m (fig. 2).

La dimension de 42 m se justifie par la décision prise au début de l'étude de concevoir la construction de manière à permettre l'introduction et l'extraction de tous les équipements, y compris un générateur de vapeur monobloc, après le complet achèvement des travaux de gros œuvre. Une porte de dimension appropriée est prévue dans l'enceinte à cet effet.

L'enveloppe intérieure est recouverte sur sa face interne d'une peau d'étanchéité métallique de 6 mm d'épaisseur. L'étanchéité de l'enveloppe extérieure est assurée par peinture et par un cuvelage du type bitumeux empêchant les infiltrations en provenance du sol. Les

enveloppes sont solidaires du même radier parcouru par un réseau de drains en communication avec l'espace entre enveloppes. Lors des épreuves de réception des enveloppes intérieure et extérieure sous les pressions effectives respectives de $3,1 \text{ kgf/cm}^2$ et $-0,013 \text{ kgf/cm}^2$, les fuites maximales admises par jour sont respectivement de $1/1000$ et $1/100$ des masses gazeuses contenues.

Les pénétrations étanches au travers de l'enceinte répondent aux mêmes principes de conception et d'essais qu'à la centrale de Doel à la différence toutefois que les ancrages sont effectués dans l'enveloppe intérieure. Les compensateurs de dilatation dont sont munies les pénétrations au travers de l'enveloppe extérieure assurent la complète indépendance des enveloppes au-dessus du radier.

Le confinement s'étend à un local auxiliaire accolé à l'enceinte du réacteur; les équipements des circuits de recirculation, d'injection de sécurité et d'aspersion extérieurs à l'enceinte du réacteur y sont montés. Les dispositions sont prises, notamment par le report des organes de commande et des moteurs à l'extérieur du local, pour que les fonctions puissent être conservées en cas d'inondation.

Pour le calcul des structures internes, on a notamment tenu compte des effets des séismes spécifiques au site de Tihange, des réactions résultant de la rupture d'une tuyauterie primaire et des pressions différentielles s'établissant dans chacune des cellules selon le point où se produit la fuite. Pour répondre à ce dernier effet, des orifices de décompression ont été prévus dans les cellules; le choix de leur orientation et de leur dimension est toutefois limité par le respect des fonctions de protection contre les projectiles et de blindage biologique de ces cellules.

Des dispositifs de maintien des équipements primaires limitent les déplacements en cas de rupture du circuit primaire et évitent la rupture des connexions des circuits de sauvegarde. Les tuyauteries de ces circuits sont calculées de manière à supporter sans dommage un déplacement de 75 mm de leur extrémité dans le sens le plus défavorable.

Le rabattement de la pression dans l'enceinte après accident majeur est assuré par un système redondant d'aspersion comprenant notamment quatre circuits de recirculation à mi-débit dont deux situés entièrement dans l'enceinte.

La filtration de l'iode qui s'échapperait de l'enveloppe intérieure en cas d'accident est assurée dans l'espace entre enveloppes; cette situation présente divers avantages:

- les conditions d'ambiance différentes de celles de l'accident et la constance de l'efficacité due à l'absence d'humidité
- la faible dimension due au fait que seul l'iode s'échappant de l'enceinte primaire doit être filtré
- le fait important qu'aucune protection contre l'incendie dû à l'échauffement n'est nécessaire.

Le circuit utilisé (fig. 2) comporte une filtration en circuit fermé et une filtration d'extraction maintenant l'espace entre enveloppes sous dépression.

La filtration en circuit fermé traite en deux heures le volume de l'espace entre enveloppes. La filtration d'extraction est assurée par deux ventilateurs de $1700 \text{ m}^3/\text{h}$ branchés à l'aval des batteries de filtration en circuit fermé.

3.3. Dispositions particulières

Diverses dispositions de nature à augmenter la sécurité d'alimentation des auxiliaires ont été prises. Des interrupteurs enclencheurs sont insérés entre chaque alternateur et son transformateur élévateur. Outre son rôle de coupler et découpler la machine sur le réseau, cet interrupteur permet de maintenir la même configuration d'alimentation des auxiliaires à tous les régimes de la centrale, supprime la nécessité de déplacer les sources lors des manœuvres de démarrage et d'arrêt et assure aux auxiliaires une source extérieure supplémentaire en cas de déclenchement des turbogroupes.

D'autre part une séparation galvanique est assurée entre jeux de barres normaux et jeux de barres secourus par l'intermédiaire des transformateurs à deux enroulements secondaires de 6 kV. Le premier enroulement alimente le réseau des auxiliaires normaux de puissances unitaires relativement élevées; le deuxième enroulement, de moindre puissance, alimente les auxiliaires secourus et permet l'utilisation d'appareillages et de liaisons de moindre calibre et de plus grande fiabilité.

En dernier lieu nous signalerons la machine de transfert du combustible. De conception originale, elle permet d'effectuer les trois opérations de basculement, translation et basculement requises pour transférer le combustible sans interruption de mouvement et à l'aide d'un seul mécanisme d'entraînement.

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STATUS OF UNITED STATES ATOMIC ENERGY COMMISSION WATER-REACTOR SAFETY RESEARCH AND DEVELOPMENT PROGRAMS

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Abstract-Résumé-Аннотация-Resumen

STATUS OF UNITED STATES ATOMIC ENERGY COMMISSION WATER-REACTOR SAFETY RESEARCH AND DEVELOPMENT PROGRAMS.

The USA continues to devote significant effort to nuclear-power-plant safety consistent with the need to increase nuclear-system size and power density and to site plants nearer load centres. The effort is directed toward providing the analytical techniques and associated experimental information that will materially increase the confidence level of reactor safety assessment in the siting, design, construction, and operation of nuclear-reactor systems. The current program emphasizes (a) accident prevention, (b) engineered safety system investigations (ESSI), (c) fuel-cladding-failure studies, and (d) development of experimentally verified analytical techniques. The products of these programs lead to the development and application of comprehensive reactor standards and codes.

The experimental accident-prevention programs result in improved nuclear-system designs by providing information on pipe rupture, fatigue and corrosion, heavy section steel technology, and anti-seismic design criteria. In the ESSI program, experiments such as those on full-length emergency cooling heat transfer (FLECHT), for studying emergency core cooling, and semi-scale blowdown are providing data for development and verification of separate analytical techniques. Assessment of the overall loss-of-coolant accident (LOCA) analysis requires interactions of blowdown, core thermal response, containment-atmosphere behaviour, and fission-product release and transport, which will be provided in integrated LOCA tests with the Loss of Fluid Test (LOFT), a 55-MW pressurized-water reactor. Fuel-cladding-failure studies provide such information as fuel-failure causes and effects and fuel-failure propagation. Data from the Power-Burst Facility in 1971 will complement data provided from previous studies at the Special Power Excursion Reactor Test-Capsule Drive Core, the Transient Reactor Test, and Oak Ridge National Laboratory. Development of the overall analyses consists in synthesizing and integrating comprehensive predictive codes for (a) system transient hydraulic/mechanical behaviour, (b) core nuclear, thermal, and mechanical response, (c) emergency-core-cooling performance, (d) containment response, and (e) fission-product release and transport. The accomplishments of the safety program are evidenced in the improved measurability in the margins of safety for reactors (which margins are the real indices of accident potential) and in the accident consequences of nuclear systems.

ETAT ACTUEL DES RECHERCHES CONCERNANT LA SECURITE DES REACTEURS A EAU DE L'USAEC ET PROGRAMME DE DEVELOPPEMENT.

Les Etats-Unis d'Amérique continuent à consacrer d'importants efforts au développement de la sécurité des centrales nucléaires, imposé par la nécessité d'augmenter leur taille et leur puissance et de les implanter près des centres de consommation. Les efforts tendent principalement à obtenir les techniques d'analyse et les informations expérimentales associées qui augmenteront sensiblement le niveau de confiance dans la sécurité du réacteur à chaque stade de la localisation, du projet, de la construction et du fonctionnement. Le programme actuel met l'accent sur: a) la prévention des accidents; b) les recherches sur les dispositifs de sécurité (ESSI); c) les études des défauts dans le gainage du combustible; d) le développement de techniques analytiques vérifiées expérimentalement. Les résultats de ces programmes conduisent au développement et à l'application de spécifications et de règles globales pour les projets de réacteur.

Le programme expérimental de prévention des accidents consiste en une meilleure conception des systèmes nucléaires tenant compte des informations acquises sur la rupture, la fatigue et la corrosion des canalisations, la technologie de l'acier sous de larges sections, et les critères antiséismiques. Dans le programme ESSI, des expériences portant sur le transfert d'urgence de la chaleur au réfrigérant sur toute la longueur du réacteur (FLECHT), le refroidissement d'urgence du cœur et une dépressurisation limitée fournissent des données pour le développement et la vérification de techniques analytiques séparées. L'analyse globale des accidents par perte de réfrigérant (LOCA) implique des interactions avec la dépressurisation, la réponse thermique du cœur, le comportement de l'atmosphère dans l'enveloppe, le dégagement et le déplacement de produits de fission, qui seront obtenues par des expériences intégrées LOCA réalisées à l'aide d'un réacteur à eau sous pression de 55 MW, le Loss of Fluid Test (LOFT). L'étude des défauts apparaissant dans le gainage du combustible fournit des informations sur leurs causes, leurs effets et leur propagation. Les données obtenues à l'aide du Power-Burst Facility (PBF) en 1971 compléteront les résultats fournis par des études antérieures effectuées à l'aide des réacteurs Special Power Excursion Reactor Test-Capsule Drive Core (SPERT/CDC), Transient Reactor Test (TREAT) et à l'Oak Ridge National Laboratory. Le développement des analyses globales a pour but la synthèse et l'intégration de règles étendues prévues pour a) le comportement en régime transitoire du système mécanique et hydraulique, b) la réponse mécanique, thermique et nucléaire du cœur, c) le fonctionnement du système de réfrigération d'urgence du cœur; d) la réponse de l'enveloppe; e) le dégagement et le transfert des produits de fission. Les réalisations du programme de sécurité sont mises en lumière par la possibilité de mesures améliorées des marges de sécurité des réacteurs (qui sont les indices réels des possibilités d'accident) et des conséquences d'accidents des systèmes nucléaires.

СОСТОЯНИЕ ПРОГРАММ КОМИССИИ ПО АТОМНОЙ ЭНЕРГИИ СОЕДИНЕННЫХ ШТАТОВ АМЕРИКИ В ОБЛАСТИ ПРОВЕДЕНИЯ НАУЧНО-ИССЛЕДОВАТЕЛЬСКИХ И ОПЫТНО-КОНСТРУКТОРСКИХ РАБОТ ПО БЕЗОПАСНОСТИ ВОДО-ВОДЯНЫХ РЕАКТОРОВ.

В США постоянно ведется работа по изучению безопасности ядерных энергетических установок, обусловленная необходимостью увеличивать размеры и энергонапряженность ядерных систем, а также располагать эти установки вблизи центров потребления энергии. Усилия направлены на разработку аналитических методов и на получение экспериментальных данных, которые существенно повысят уровень надежности в оценке безопасности реакторов при выборе площадки, а также при проектировании, сооружении и эксплуатации реакторных систем. Современная программа по безопасности включает: а) предотвращение аварий, б) инженерные исследования систем безопасности, в) изучение повреждений оболочек топливных элементов, г) разработку и экспериментальное опробование аналитических методов. Результаты выполнения этих программ являются основой для разработки и использования исчерпывающих реакторных стандартов и норм.

Программы экспериментальных исследований по предотвращению аварий имеют целью разработку усовершенствованных конструкций ядерных систем и получение информации, касающейся повреждения трубопроводов, усталости и коррозии материалов, технологии изготовления тяжелых стальных секций и антисейсмических конструктивных критериев. По программе инженерных исследований безопасности систем проводятся эксперименты с полномасштабной аварийной системой охлаждения в целях изучения аварийное расхолаживание активной зоны и с полумасштабной системой продувки. Полученные результаты будут использованы для разработки и экспериментального опробования отдельных аналитических методик. Оценка результатов анализа, проведенного по последствиям аварии при полной потере теплоносителя, требует согласования параметров продувки, тепловых характеристик активной зоны, характеристик атмосферы внутри защитной оболочки, а также данных о выделении и распространении продуктов деления. Эти аспекты будут исследованы при комплексных испытаниях с полной потерей теплоносителя (LOFT) на реакторе с водой под давлением мощностью 55 МВт. Исследование повреждений топливных элементов и оболочки позволит получить данные как о причинах повреждения топливных элементов, так и о влиянии распространения этих повреждений. Данные, полученные в 1971 году, на установке для изучения импульсов мощности, дополняют результаты, полученные в предыдущих исследованиях при проведении специальных испытаний на выброс радиоактивности на энергетическом реакторе с подвижной активной зоной, на реакторе по исследованию переходных процессов, а также в Ок-Риджской национальной лаборатории. Разработка методов общего анализа заключается в синтезировании и обобщении всеобъемлющих правил для: а) проведения системы при гидравлических и механических переходных процессах, б) ядерных, тепловых и механических поведений активной зоны, в) действия аварийной системы охлаждения активной зоны, г) действия защитной оболочки и д) выделения и перевозки продуктов деления. Завершение работ по программе исследований вопросов безопасности позволит получить более точные данные о коэффициентах безопасности для реакторов (коэффициенты безопасности которых реально определены для потенциальных аварий), а также о последствиях аварий на ядерных установках.

ESTADO DE LOS PROGRAMAS DE LA USAEC DE INVESTIGACION Y DESARROLLO EN MATERIA DE SEGURIDAD DE REACTORES DE AGUA.

Los Estados Unidos de América continúan prestando gran atención a la seguridad de las centrales nucleares, dada la necesidad de aumentar la capacidad y densidad de potencia de los reactores, así como de emplazar las centrales más cerca de los centros de consumo. Los trabajos se encaminan hacia la consecución de técnicas analíticas y datos experimentales conexos que permitan elevar considerablemente el grado de confianza en la evaluación de las condiciones de seguridad que ofrece el emplazamiento, proyecto, construcción y funcionamiento de los reactores nucleares. El programa actual hace hincapié en los siguientes temas: a) prevención de accidentes; b) investigaciones sobre sistemas de seguridad por medios técnicos (ESSI), c) estudios sobre roturas de vainas del combustible y d) perfeccionamiento de técnicas analíticas comprobadas experimentalmente. Los resultados de estos programas permiten elaborar y poner en práctica normas y reglamentos detallados referentes a los reactores.

Gracias a los programas experimentales de prevención de accidentes es posible mejorar las características de los sistemas nucleares, pues dichos programas proporcionan conocimientos sobre las causas de la rotura, fatiga y corrosión de las tuberías, la tecnología de los grandes perfiles de acero, y las condiciones que requiere el proyecto de reactores asísmicos. Dentro del programa ESSI, por medio de experimentos tales como los de transmisión de calor en toda la longitud para estudiar la refrigeración del núcleo en caso de emergencia (FLECHT) y los de pérdida parcial del agua refrigerante (Semiscale Blowdown), se están obteniendo datos para el perfeccionamiento y verificación de técnicas analíticas independientes. A fin de juzgar los resultados del análisis global del accidente de pérdida del refrigerante (LOCA) es preciso conocer los efectos combinados del escape parcial de agua, de la reacción térmica del núcleo, del comportamiento de la contención y el medio circundante, así como de la dispersión y transporte de los productos de fisión, efectos que se estudiarán en ensayos LOCA integrales a realizar en la instalación experimental Loss of Fluid Test (LOFT), un reactor de agua a presión de 55 MW. Los estudios sobre roturas de vainas del combustible proporcionan información tal como la referente a las causas y efectos de las roturas del combustible y su propagación. En 1971, los datos obtenidos con la Power-Burst Facility (instalación para el estudio de grandes aumentos bruscos de potencia) completarán los proporcionados por estudios anteriores en el Special Power Excursion Reactor Test-Capsule Drive Core (Reactor experimental para el estudio de saltos de potencia - Núcleo con cápsulas móviles), el Transient Reactor Test (Reactor experimental para el estudio de perturbaciones transitorias) y en el Laboratorio Nacional de Oak Ridge. La finalidad de los análisis globales es preparar y sistematizar reglas detalladas para predecir: a) el comportamiento transitorio de los sistemas, desde el punto de vista hidráulico y mecánico; b) el comportamiento nuclear, térmico y mecánico del núcleo, c) el funcionamiento de la refrigeración de emergencia del núcleo, d) el comportamiento de la contención; e) el escape y transporte de los productos de fisión. Los progresos del programa de seguridad se evidencian por el hecho de que es posible determinar con más exactitud los márgenes de seguridad de los reactores (márgenes que son en realidad los índices de probabilidad de accidentes) y las consecuencias de los accidentes en los sistemas nucleares.

INTRODUCTION

The Water-Reactor Safety Program in the United States is being conducted in accordance with the Water-Reactor Safety Program Plan, WASH-1146, issued in February 1970. This plan was produced by the Water-Reactor Safety Program Office, operated by Idaho Nuclear Corporation at the National Reactor Testing Station. The draft plan was extensively reviewed by the appropriate elements of the nuclear community including Regulatory, the Advisory Committee on Reactor Safeguards, the commercial nuclear industry, and the AEC's laboratories and principal contractors, and was issued reflecting a consensus of the comments, thus forming a generally agreed-upon approach to proceeding with the technical effort. The plan includes priority considerations and makes recommendations concerning those parts of the program which would be appropriate for the industrial sector to perform as distinguished from those which are more appropriate to be performed as AEC programs.

In perspective, it should be emphasized that much of the safety development work described in the plan represents attention to a number of levels of protection in addition to the normal emphasis which the plant designers, constructors, and operators must give to producing and operating a well engineered, reliable and safe reactor plant. As a result of these considerations, principal emphasis in the program is directed toward the technical basis for assuring reactor plant reliability and integrity through a systematic engineering approach embodying the development and application of improved engineering standards, codes and criteria and strengthened quality assurance practices. This effort is strongly reinforced by regulatory emphasis on compliance with appropriate codes, criteria and standards, for example the recently issued Quality Assurance Criteria for Nuclear Power Plants.

At the same time, it has been recognized that it may not be possible at this stage of development of nuclear power to obtain, under all circumstances, the specific engineering features or quality assurance which will guarantee the necessary primary system integrity over the life of the plant. Accordingly, a substantial effort is devoted to the development and testing of special engineered safety features which can interrupt the course of a failure or accident at an acceptable safety level or can ameliorate the consequences of an accident in the unlikely event one occurs.

The fundamental objectives of the Water-Reactor Safety Program are to provide the AEC and industry with (a) the safety-related information required to develop and regulate the safe application of water reactors to power production by performing the necessary research, development and testing, and (b) an integrated procedure by which the industrial and public safety interests can be co-ordinated.

Reactor safety must be an integral part of design, construction, maintenance and operation. The activities undertaken in the safety program cannot, of themselves, assure the safety of reactors, but must be more properly thought of as a superposition on activities which must be undertaken by the reactor industry to assure that operable, reliable and safe reactor plants with a high degree of availability and economy are the result of its efforts. Within this perspective the safety program provides information needed for the understanding of the potential consequences resulting from abnormal operation or accidents so that adequate design, construction and operation requirements and procedures can be formulated.

The prime concern of reactor safety is to ensure the control of radioactive material produced by operation of the reactor. Independent barriers to fission product escape, such as the fuel element and cladding, the primary system, and the containment are provided so that, generally, multiple failures must occur to pose a threat to the safety of the general public. To understand how accidents interact with these barriers, it is convenient to subdivide the program into the following major problem areas, which are listed in order of importance: (a) accident prevention, (b) emergency core cooling, (c) containment, and (d) control of fission products. The following description of the status and scope of the program, and recent accomplishments is presented in accordance with this framework.

1. ACCIDENT PREVENTION

In the USAEC Water-Reactor Safety Program, codes and standards are the vehicle for implementing a disciplined engineered approach to accident prevention. Codes and standards are being developed in the program for application to all facets of reactor design, fabrication, and operation. It is through codes and standards, and adequate quality control that primary system integrity is optimized and control is exercised over the factors which produce accidents. Safety assessment work in the United States includes development of techniques for primary system inspection and methods for analysis of reliability and modes of failure. As these techniques are developed, they are reported and disseminated for use by the reactor designers and operators.

1.1. Studies of factors affecting primary system integrity

The factors that affect the integrity of the primary system boundary include: (a) Materials of construction and methods of fabrication, (b) Mechanics of fracture including pipe rupture, and (c) Fracture initiating mechanisms.

Studies of materials and of fabrication methods include: (a) Experimental and analytical investigations of nuclear vessel nozzle connections [1, 2], in order to develop and validate analysis methods for predicting stress and strain around nozzles on spherical or cylindrical shells. (b) Engineering investigations of the physical and mechanical properties of the heavy section steel (greater than 6 inches thick) used in large nuclear reactor pressure vessels [3, 4]. These latter investigations constitute the Heavy Section Steel Technology (HSST) Program, needed for developing design and operational criteria for thick-section vessels.

Phenomena being studied include fracture mechanics, fatigue and crack propagation, and materials transition temperatures with variables being heat treatment, extent of materials imperfections, irradiation, complexity of stress states, and composition of materials. Simulated service tests with large vessels will conclude the program.

The fracture mechanics studies include the following: (a) Experiments being performed to determine conditions for initiation of ductile rupture for full-scale BWR and PWR pipe sections [5, 6] which to date have shown that crack extension or arrest can be predicted to within 6% or better. (b) A primary coolant piping rupture study [7] is being performed to provide information for assessing the in-service integrity of BWR piping related to low-cycle material fatigue under reactor operating conditions, and information regarding reliability, fracture mechanics and stresses in complex geometries is also being derived. (c) A study to provide information on the effect of long-term irradiation on the fracture toughness of heavy section pressure vessel steels [8] is being performed.

Fracture initiation studies on stainless steels [9] are being directed principally at determining the conditions causing stress corrosion cracking. The initial tests in the program are emphasizing high stress fatigue conditions and sensitized material simulating weld heat affected zones and nozzle extensions which have received stress relieving heat treatments.

1.2. Development of techniques for assessing system integrity

In addition to detection of leaks and flaws by conventional methods, flaws can be located by means of ultrasonic noise emitted during stressing. The AEC has conducted programs to provide an understanding of acoustic emission and transmission from flaws in stressed pressure envelopes [10, 11]. Diagnostic systems based on this approach have been successfully tested on large pressure vessels. The tests have demonstrated the capability of the systems for detecting and locating flaws at stress levels that are a small fraction of the design yield stress of the vessel.

1.3. Development and application of standards

The purpose of the AEC standards program is to develop standards for reactor plant design, analytical methods, materials, fabrication, construction, quality assurance, inspection and operation [12].

This program results from the recognized need to achieve a predictable level of quality required to assure safe, reliable operation of nuclear plant equipment. Standards provide the primary means of achieving this objective by documenting for uniform application requirements resulting from experience and advances in technology.

Many standards have already been prepared under this AEC program, and are currently being applied on the major AEC reactor and test facility engineering projects. Through this process, the standards will be improved and validated, and will provide a base of successful experience which can be applied with confidence to future nuclear programs.

2. EMERGENCY CORE COOLING

2.1. Discussion of postulated accident sequence and program rationale

The loss-of-coolant accident resulting from the postulated failure of a large primary coolant pipe is used as a basis for reactor safety assessment and emergency core coolant system design. In this accident sequence rapid decompression occurs as the coolant discharges through the rupture. The resulting expansion and voiding of the water in the core and/or reactor scram terminates production of fission energy leaving only sensible heat in the fuel and continuing fission product decay as heat sources. Thermal transients with fuel and cladding follow the loss of core coolant that results from system rupture. The first clad temperature transient, up to about 2000°F, is due primarily to the redistribution of sensible heat stored in the fuel pin. The maximum temperature obtained depends mainly upon the amount of stored energy removed during discharge of the coolant.

Because of continued heating by fission product decay the fuel and cladding temperature begins to increase after blowdown is over unless some source of supplemental core cooling is provided. If the temperature reaches approximately 2000°F, chemical reactions between the cladding and water steam mixture may occur, possibly leading to rapid cladding oxidation, additional energy generation and embrittlement of the cladding.

Depending on the temperature reached, the cladding could crumble due to thermal stresses induced by the cooler emergency core cooling water, possibly resulting in partially blocking some of the cooling channels. Without cooling, fuel could eventually become molten and under extreme postulations could ultimately lead to failure of the containment structure. Such extreme postulations are not considered credible, but nevertheless have been examined in the course of designing power reactors and engineered safety features such as emergency core cooling systems.

The chain of events comprising the loss-of-coolant accident will be interrupted before there is significant damage to the core if a supplementary supply of water sufficient to remove stored heat and decay heat from the fuel is supplied to the core. The emergency core cooling system is thus an extremely important adjunct to a water reactor to accommodate unlikely primary system failures which are beyond the range of capability of normal make-up water systems. The USAEC program on water-reactor safety includes extensive efforts to develop a basis for the proper design and assessment of these emergency core cooling systems.

The large size and cost of water-reactor plants preclude full-scale tests to study accident behaviour or the effectiveness and reliability of safety systems under accident conditions. Hence analytical models whose conservatism has been assessed by comparison of analytical predictions with data from smaller scale experiments provide a provisional basis for understanding the accident and the effectiveness of the safety systems. A summary of the status of analysis development and experimental programs for characterizing the loss-of-coolant accident follows. The overall adequacy and maturity of the analysis will be confirmed in the LOFT integral nuclear experiments to be described later.

2.2. Hydraulic behaviour during a loss-of-coolant accident

Analytical methods describing fluid behaviour and conditions during the decompression process have been developed, including a computer code for predicting the hydraulic forces occurring during the subcooled portion of the blowdown [13], and a computer code [14] for predicting transient fluid behaviour during both the subcooled and saturated blowdown. Methods for describing fluid behaviour during reflooding with emergency core cooling water are under further development.

Data from blowdown tests using a vessel with stagnant fluid conditions and conducted in the Containment Systems Experiment [15] at Hanford have been applied in the determination of exit nozzle discharge coefficients. Decompression experiments have recently been performed and further work is planned in the LOFT Semiscale Blowdown System [16]. This system, 10 ft³ in volume, included a single circulating loop, a pump, steam generator, electrically heated core of 1.2 MW and a coolant injection system. These tests are conducted to help confirm and guide the hydraulic analysis development effort.

2.3. Core thermal behaviour

Two types of core thermal analysis methods have been developed for describing core thermal behaviour, the principal concern during a loss-of-coolant accident. They are: (a) Single-rod single-fluid channel codes

[17, 18] employing empirical heat-transfer correlations [19] and providing capability for detailed calculation of core heat removal. (b) A three-dimensional model [20] with the capability for calculating in large bundle arrays, spatial energy redistribution, rod thermal history, and energy removed by the fluid.

Limited information on core heat transfer during blowdown indicates that the high heat transfer coefficients typical of nucleate boiling extend further into blowdown than is predicted by steady-state correlations [16]. Heat transfer during the latter phases of the accident has been studied in multipin arrays [20], where energy transfer by radiation and convection have compared favourably with the three-dimensional code predictions.

Heat transfer coefficients during reflooding with the emergency core cooling water have been obtained from the Full-Length Emergency Cooling Heat Transfer (FLECHT) Test Project [21 - 24]. Full-length bundles of stainless-steel or Zircaloy-clad electrically heated fuel rod simulators were employed in engineering parametric tests designed to provide data on important variables which affect the thermal response of the rods. The tests covered a wide spectrum of coolant delivery conditions for both pressurized and boiling water reactors.

2.4. Structural behaviour studies

During blowdown following a large break, hydraulic loads well beyond those which are encountered in normal operation must be accommodated. A system analysis code [25] has been developed for predicting the forces and accelerations generated during fluid decompression on structural members, reactor internals, and reactor primary systems. The code can accommodate seismic excitation together with loss-of-coolant loads. Predictions [16] of thrust forces, accelerations, longitudinal and circumferential strains obtained with this code have compared well with the values measured in the semiscale experiments.

2.5. Fuel failure investigations

The thermal response of a reactor core under abnormal conditions might be significantly affected by various types of fuel failure such as distortion of the fuel pins, cladding swelling and rupture, cladding embrittlement, fuel pin failure propagation, and fuel slumping and migration. The potential for fuel failure exists whenever there is mismatch between power level and available cooling such that cladding and fuel temperatures exceed design limits. Loss-of-coolant accidents and the reactivity accidents represent two extreme conditions which can lead to fuel failure. Fuel performance limits determine safety margins and provide the bases for establishing operating limits and design criteria. Fuel failure can be held to acceptable levels by proper design and operation of reactors and their associated engineered safety systems.

Fuel failure during reactivity transients has been studied in closely related programs conducted by the Argonne National Laboratory in the TREAT [26] reactor and by the Idaho Nuclear Corporation in the SPERT IV Capsule Driver Core (CDC) [27, 28]. Reactivity induced fuel failure thresholds and effects are currently receiving little attention because of the design features which have been incorporated into water reactors to

prevent significant inadvertent reactivity additions. Fuel failure under loss-of-coolant accident conditions has been studied at Oak Ridge National Laboratory [29] in out-of-pile burst tests on cladding representative of that used in both PWRs and BWRs. In-pile tests have been conducted in TREAT on clusters of Zircaloy-clad fuel rods. Both programs were conducted to determine whether swelling and rupture interfere with the effectiveness of Emergency Core Cooling (ECC). Test results have shown (a) that significant ballooning can occur without significantly impairing the effectiveness of emergency core cooling, and (b) that metal water reaction induced embrittlement in zirconium cladding must be considered depending upon the extent of clad oxidation.

The Power Burst Facility (PBF), to be completed this year, will extend the range of fuel failure investigation to conditions more representative of a reactor environment. The PBF facility [30] consists of a reactor with a flux trap containing a large in-pile tube in which the test fuel is placed. Associated with the in-pile tube is a coolant loop system which will provide typical reactor system operating conditions. The PBF facility can produce a steady-state power density of about 28 kW/ft in a cluster of nine fuel rods. Also the power can be programmed to simulate decay heat in unirradiated test fuel.

The PBF program will include power-cooling-mismatch tests to identify fuel failure thresholds and potential for propagation, resulting from loss-of-flow, flow blockage, local hot-spot generation and reactivity transients. The effects of fuel burnup and degraded Emergency Core Cooling System (ECCS) performance on the thresholds and consequences of fuel failure will also be investigated.

2.6. LOFT integral tests

The LOFT Program [31] of integral nuclear experiments being planned at the National Reactor Testing Station is viewed as the culminating test series for confirming the ability of analysis to predict reactor response to a loss-of-coolant accident arrested by ECC. The experiments to be performed in a 55 MW(th) pressurized water reactor will incorporate all interacting events, coupled in proper time sequence, that may occur in a large PWR loss-of-coolant accident.

The LOFT facility provides a single primary coolant loop to represent the unbroken loops of a large PWR and a blowdown loop to represent the ruptured loop. The blowdown loop contains pressure drop devices for simulating major components such as the pump and steam generator. Rupture devices located in the blowdown loop permit simulation of large or small inlet or outlet piping breaks. The LOFT cores and the emergency core cooling systems will be designed to be representative of those in large PWRs.

The first LOFT tests, scheduled for late 1973, will be non-nuclear experiments to test the plant and its blowdown response and will progress to a nuclear test representing double-ended failure of the inlet pipe to the reactor vessel. This latter test is a major objective of the test series since analysis has shown these conditions to place severe demands on emergency core cooling systems. Later tests will include degraded ECCS operation.

3. CONTAINMENT

Analytical methods [32, 33] have been developed for predicting the temperature and pressure history of the containment atmosphere following a loss-of-coolant accident.

Confirmatory tests for PWR containment have been conducted at the Carolina-Virginia Tube Reactor plant [34].

The test data indicated the following: (a) Measured values for condensing steam heat transfer coefficients were a factor of three to five larger than those predicted by the correlations which had been used in the code [32]. (b) Where the experimental heat transfer coefficients were used, code predictions [32] showed reasonable agreement with the temperature and pressure histories. (c) The effectiveness of the containment spray for reducing pressure and temperature agreed with predictions. (d) The test data indicated that predicted values for peak pressure are conservative.

4. CONTROL OF FISSION PRODUCTS

The acceptability of sites for nuclear power reactors is based, in part, on a conservative assessment of potential consequences (radiation doses) resulting from the release of radioactive material from the reactor. In order to aid reactor licensees by providing a uniform method for determining the acceptability of reactor sites, the AEC developed a conservative model which can be used as guidance in determining reactor site acceptability. This model is defined in 10 CFR Part 100 of the Commission's regulations and in document TID 14844 referenced in Part 100. Recently the Commission has developed additional and more specific guidance (Safety Guides 3 and 4) on assumptions which are acceptable to the AEC for evaluating the consequences of a loss-of-coolant design basis accident.

The water-reactor safety research and development programs of the Commission and the nuclear industry have attempted to correlate with and/or determine the degree of conservatism of developed models. In a loss-of-coolant accident in which the engineered safety features function before there is extensive cladding failure and fuel melting, the volatile fission products which have accumulated in the plenum or gas space of the fuel pin could be released from perforations of the fuel pin cladding. Research and development studies have evaluated the amount and kind of fission products which could be in the gas space and have correlated experimental data with analytical work [35 - 41].

Since the fuel cladding is only one barrier against fission product release to the environment, calculations and correlation with experiments are being made to evaluate the transport of the fission products from the core region through the primary system and ultimately into and out of the containment structures. Fission products present in the containment atmosphere are then available for release into the environment since the containment structure is the last step in the defense-in-depth concept of fission product retention. For this reason fission product transport and depletion within containments have been extensively

studied. For example, models exist for predicting the rate of depletion of molecular iodine and particulate material from the containment atmosphere by both natural processes and engineered systems [42, 43]. Further, experimental data show that with the exception of noble gases and certain chemical forms of iodine, fission products can be removed rapidly and effectively from containment atmospheres by spray systems employing chemical additives [43]. Filter systems are also effective for removal of all species except noble gases and certain chemical forms of iodine. However, rates of removal for filter systems are lower than for sprays and are limited by the rate at which the containment atmosphere can be circulated through the filter systems containing the media for fission product retention.

The chemical forms of iodine that are difficult to remove with either spray or filter systems have been identified as organic iodides (principally methyl iodide) and hypiodous acid [44, 45]. While early work showed that up to 10% of the iodine appearing from heated fuel specimens could be present in forms difficult to remove [46], recent examination of the information available from experiments has shown that the fraction of iodine present as persistent species depends upon the mass level of iodine in the containment atmosphere. Fractions as low as 0.001 have been observed for iodine mass concentrations in the containment expected following an accident (10 000 to 100 000 $\mu\text{g}/\text{m}^3$) [47].

CONCLUSIONS

The USAEC Water-Reactor Safety Program is devoting a significant effort to nuclear power plant safety through emphasis on assurance of plant integrity, operational reliability, and adequacy in safety system performance. Implementation of the effort is according to the Water Reactor Safety Program Plan, which subdivides the program into investigations on accident prevention, emergency core cooling, containment, and control of fission products.

In the area of accident prevention the assurance of plant integrity and operational reliability is derived through disciplined formulation and rigorous application of codes and standards for plant design, construction, operation and maintenance. Primary system integrity research and development is emphasizing investigations of materials behaviour and fabrication methods, mechanics of materials failure, and fracture initiating mechanisms. These studies form the basis for improved criteria in codes and standards and hence improved system reliability.

In the area of emergency core cooling, containment, and control of fission products, investigation of these engineered safety systems, which are provided as part of the reactor system design to mitigate the consequences of unlikely accidents, form a significant fraction of the US Water-Reactor Safety Research and Development effort. These studies have emphasized the development of analytical methods for describing the accident sequence and evaluating the adequacy of emergency safety systems. Also emphasized are comprehensive supporting experiments which provide data for confirming the capability of the analysis to predict the performance of safety systems.

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SAFETY OF LIGHT-WATER REACTORS DESIGNED IN THE FEDERAL REPUBLIC OF GERMANY

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Abstract—Résumé—Аннотация—Resumen

SAFETY OF LIGHT-WATER REACTORS DESIGNED IN THE FEDERAL REPUBLIC OF GERMANY.

Because of the comparatively high population density in the Federal Republic of Germany, both suppliers of light-water reactors, AEG-Telefunken and Siemens AG., who co-operate in the Kraftwerk-Union AG., adapted their BWR and PWR safety principles to urban siting at an early stage. The safety features developed by both suppliers comply with current international practice in protecting the environment in the event of a rupture of the primary circuit. In addition, special stress is laid on avoiding such leakage accidents, on improving the efficiency and reliability of safeguards by systematic redundancy and diversity, and on reducing release of radioactivity during normal operation and after accidents.

The design principles and the special quality standards of PWR's and BWR's under construction in the Federal Republic of Germany are reported. As compared with international practice, it can be seen that the safety precautions against a rupture of the primary coolant system are more thorough. Further, some special details of containment design and emergency cooling systems are presented. The efficiency of gaseous and liquid waste systems is demonstrated on the basis of the release rates experienced with operating light-water reactor plants in the Federal Republic of Germany.

Finally, a survey is made of the aims of R and D programs on reactor safety either planned in co-operation with the Federal Ministry of Education and Science or performed individually by AEG and Siemens.

LA SECURITE DES REACTEURS A EAU LEGERE MIS AU POINT EN REPUBLIQUE FEDERALE D'ALLEMAGNE.

La densité de population élevée de la République fédérale d'Allemagne a amené les deux constructeurs de réacteurs à eau légère, AEG-Telefunken et Siemens, coopérant au sein de la Kraftwerk-Union, à adapter depuis longtemps la conception, au point de vue de la sécurité, de leurs réacteurs à eau sous pression et à eau bouillante aux exigences posées par l'érection de centrales nucléaires dans des centres urbains. Les deux constructeurs ont ainsi mis au point des conceptions qui sont non seulement conformes aux mesures appliquées internationalement en vue de la protection du milieu en cas de rupture du circuit primaire, mais qui visent surtout à prévenir de telles ruptures, par la redondance systématique et la diversité des moyens mis en œuvre pour assurer la sécurité et par leur efficacité. En outre la protection de la population avoisinante contre la libération de produits radioactifs en période normale et en cas d'accident a fait l'objet d'une attention toute particulière.

Les réacteurs à eau sous pression et à eau bouillante actuellement en construction en République fédérale d'Allemagne fournissent des exemples de procédés de construction et de standards de qualité spéciaux qui augmentent encore la sécurité contre une rupture du circuit primaire par rapport aux standards internationaux. Des détails des enceintes étanches et des systèmes de refroidissement de secours sont également présentés. L'efficacité des systèmes d'évacuation des rejets radioactifs liquides et gazeux est démontrée en prenant pour exemple les quantités d'effluents radioactifs rejetées par les centrales nucléaires à eau légère de la République fédérale en exploitation.

Enfin, le mémoire donne un aperçu des programmes d'études et de réalisations entrepris actuellement en vue de perfectionner les techniques de sécurité, soit par les deux constructeurs de réacteurs AEG-Telefunken et Siemens seuls, soit en coopération avec le Ministère fédéral de l'éducation et des sciences.

БЕЗОПАСНОСТЬ ЛЕГКОВОДНЫХ РЕАКТОРОВ, КОНСТРУИРУЕМЫХ В ФЕДЕРАТИВНОЙ РЕСПУБЛИКЕ ГЕРМАНИИ.

Ввиду относительно высокой плотности населения в Федеративной Республике Германии, оба поставщика легководных реакторов — "АЕГ-Телефункен" и "Сименс АГ", сотрудничающие в рамках фирмы "Кraftwerk Union АГ", — на ранней стадии приняли принципы безопасности для своих реакторов типа "BWR" и "PWR" применительно к условиям их размещения в городских районах. Требования по безопасности, разработанные обоими поставщиками, отвечают современным международным нормам по защите окружающей среды в случае разрыва первого контура реактора. Кроме того, особое внимание обращено на устранение возможности аварий, связанных с утечкой, на повышение эффективности и надежности систем защиты путем систематического повышения запаса прочности и многократного дублирования, а также на уменьшение утечки радиоактивности в процессе нормальной эксплуатации реакторов и в результате аварий.

Излагаются принципы конструирования и специальные требования, предъявляемые к качеству реакторов типа "PWR" и "BWR", сооружаемых в Федеративной Республике Германии. При сравнении с международной практикой можно увидеть, что меры предосторожности, предпринятые против разрыва первого контура теплоносителя, являются в данном случае более тщательными. Далее рассматриваются некоторые особые детали конструкций противоаварийных оболочек реакторов и аварийных систем охлаждения. Эффективность систем удаления газообразных и жидких отходов демонстрируется на основе фактических данных о количествах удаляемых отходов, зарегистрированных на действующих электростанциях с легководными реакторами в Федеративной Республике Германии.

Наконец, в докладе рассмотрены цели программ научно-исследовательских и опытно-конструкторских работ в области безопасности реакторов, запланированных в сотрудничестве с Федеральным министерством по вопросам образования и науки или осуществляемых самостоятельно "АЕГ" и "Сименсом".

SEGURIDAD DE LOS REACTORES DE AGUA LIGERA EN PROYECTO EN LA REPUBLICA FEDERAL DE ALEMANIA.

Debido a la elevada densidad demográfica de la República Federal de Alemania, las dos compañías suministradoras de reactores de agua ligera, AEG-Telefunken y Siemens A. G., que cooperan en la Kraftwerk-Union A. G., han adaptado desde la primera fase de planificación sus normas de seguridad relativas a los reactores de agua hirviendo y de agua a presión a las necesidades de los emplazamientos urbanos. Las características de seguridad desarrolladas por ambas compañías satisfacen la práctica internacional corriente para la protección del medio ambiente en caso de ruptura del circuito primario. Se está haciendo además mucho hincapié en evitar tales accidentes, con sus consiguientes fugas radiactivas, al mejorar la eficiencia y confiabilidad de las medidas de seguridad mediante la sistemática duplicación y diversificación de las mismas, y al reducir los escapes de radiactividad durante el funcionamiento normal o como consecuencia de un accidente.

La memoria informa sobre los principios de diseño y las normas especiales de calidad de los reactores PWR y BWR en construcción, en la República Federal de Alemania, con lo que, en comparación con la práctica internacional, se puede ver que la seguridad contra una ruptura del sistema de refrigeración primario es más completa. Se presentan además algunos detalles especiales del diseño de la contención y de los sistemas de refrigeración de emergencia. Se demuestra la eficiencia de los sistemas de eliminación de desechos líquidos y gaseosos basándose en las proporciones de escapes observadas en las centrales de reactores de agua ligera que funcionan en la República Federal.

Se informa sobre los objetivos, R y D relativos a seguridad de reactores, ya estén planeados en cooperación con el Ministerio Federal de Educación y Ciencia, o desarrollados individualmente por AEG y Siemens.

1. INTRODUCTION

In the Federal Republic of Germany, both suppliers of light-water reactors, AEG-Telefunken and Siemens, who co-operate in the Kraftwerk Union AG, adapted their BWR and PWR safety design to urban siting at an early stage. The three factors that substantially influenced the safety design of light-water reactors designed in the Federal Republic are described below.

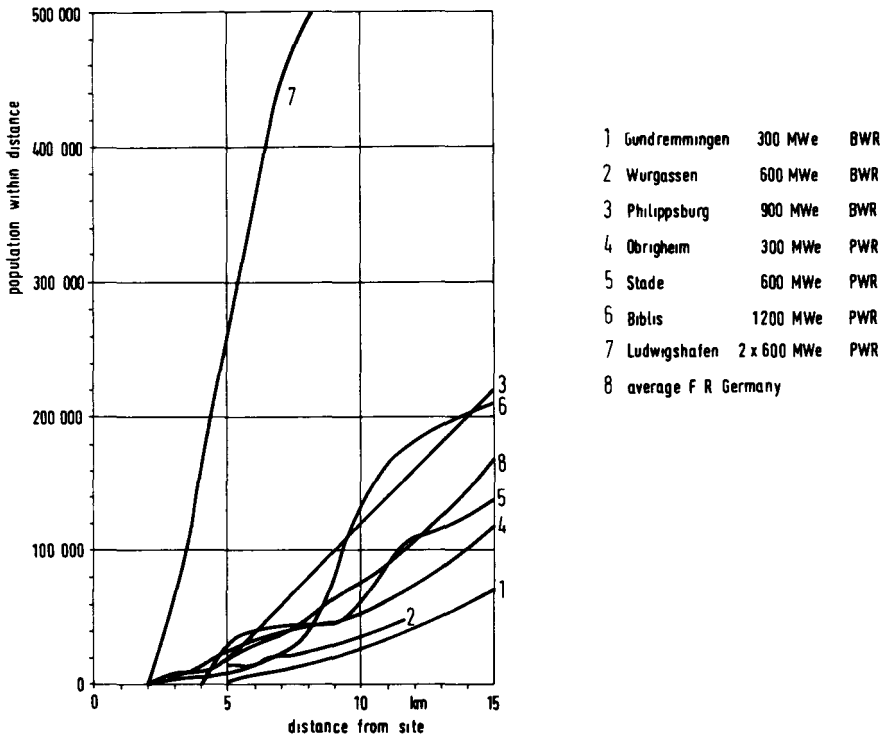


FIG. 1. Population densities in the surroundings of nuclear power plants in the Federal Republic of Germany.

(1) Germany is very densely populated; this is especially true of areas near such major rivers as the Rhine and Elbe, where the most suitable sites for large power stations are found. Moreover, there is strong incentive to build nuclear stations producing process steam and electricity for large chemical plants, which in most cases are situated within the municipal areas of major cities. Thus, as Fig.1 demonstrates, the siting of nuclear plants in densely populated areas is inevitable and urban siting is imperative for the future, not only for economic reasons, but also to reduce air pollution in highly industrialized cities.

(2) In the early stages of light-water reactor development, Siemens and AEG-Telefunken co-operated with the American pioneers in this reactor field, Westinghouse and General Electric. The late start of nuclear research in the Federal Republic of Germany — it was prohibited until 1955 — was certainly one of the reasons for this. As a consequence, both the AEG and Siemens reactor designs in general and their safety standards developed from those in the United States of America.

(3) In spite of this relatively late start, a concentrated effort enabled the German light-water reactors to reach their present stage of advanced development and technical competence. AEG and Siemens, co-operating through the Kraftwerk Union, now have (April 1971) 15 light-water reactor stations under turnkey contracts in various stages of operation, construction and planning. Four demonstration stations have accumulated twenty operating years, nine commercial stations in the range of 700 to 1300 MW(e) are being built in the Federal Republic of Germany, one of which is currently the largest nuclear power station in the world, and one each in the highly industrialized and densely populated countries of Holland and Austria. The technology of present-day commercial stations thus still relates to that in the United States, but at the same time these stations have acquired substantially different features. The constant and fruitful collaboration of German safety authorities such as the Reactor Safeguards Commission (RSK), the Institute of Reactor Safety (IRS) and the Inspectors' Organizations (TÜV) with AEG and Siemens resulted in a set of safety standards for our light-water reactors which adapt them to urban siting.

It is thus a combination of siting problems, early adaptation of United States technology, and intense development of light-water reactors for electricity supply in the Federal Republic of Germany that forms the background for the safety features reported here. In view of the space limitations, the paper deals only with safety features relating to the two major safety areas: activity release under normal operating conditions and prevention or control of primary system ruptures up to the maximum credible accident. Finally, a survey is given of the R and D program on reactor safety implemented by AEG and Siemens in co-operation with the Federal Ministry of Education and Science.

2. RADIOACTIVITY RELEASE IN NORMAL OPERATION

2.1. Gaseous release

As in all countries, the limits in the Federal Republic for release of radioactivity from nuclear plants are related in some fashion to the level of radioactivity found in nature (about 150 mrem/yr). In contrast, however, to the United States, where until recently plant licences limited exposure from gaseous release in the surroundings of a nuclear plant to a maximum of 500 mrem/yr, in the Federal Republic this limit was set for the first generation of demonstration plants to below 150 mrem/yr, and for plants currently under construction a maximum exposure of 30 mrem/yr will be the licence limit.

These strict limits forced the reactor industry to develop an improved off-gas technology. For example, for the first large demonstration plant, the 250-MW(e) BWR in Gundremmingen, which has been operating since 1966, AEG developed an off-gas system on the basis of charcoal absorption. All exhaust air from the turbine condenser is passed over a catalytic recombiner to remove free oxygen and hydrogen, and, after further drying and cooling, the remaining air is passed through columns filled with charcoal. The charcoal is specially treated for high adsorption qualities, the fission products passing slowly over the charcoal particles; the total hold-up time

is different for each fission product and may be increased by adding more columns. At Gundremmingen, the hold-up time is 14 days for xenon, at the same time all fission products are removed except ^{133}Xe and ^{85}Kr . Iodine is quantitatively absorbed. After passing through the charcoal columns, the air is released to the stack. The process decreases the activity release to the stack by a factor of 30 000 with respect to release from the reactor and by a factor of about 200 with respect to the 30-minute hold-up system commonly in use for BWR's outside the Federal Republic.

This system has operated very satisfactorily for many years in both the Gundremmingen and Lingen 250-MW(e) BWR stations; it is now being designed for a 40-day hold-up time for all commercial BWR's, reducing activity release by a factor of 3000 with respect to the 30-minute hold-up. As a result of this off-gas system, gaseous activity release has been extremely low. At the 250-MW(e) Gundremmingen BWR, over its five years of operation, the maximum exposure in the surroundings has been only a few tenths of a mrem/yr, not exceeding 0.3 mrem/yr in years with a high rate of fuel failure; in other words, even the maximum exposure in the surrounding area was not traceable and amounted to only a few thousandths of the natural radioactivity. At the 250-MW(e) BWR Lingen, the average exposure was about a factor of ten higher than at Gundremmingen, since over prolonged periods of time the off-gas system had not been operated due to generally low release rates — even for Lingen, however, the maximum exposure in the surrounding area was limited to less than 5 mrem/yr.

The gaseous release rate of the Siemens-built 320-MW(e) PWR demonstration plant Obrigheim was 6200 Ci of noble gases and 28 mCi of iodine-131 during normal operation in 1970; 1500 Ci of noble gases and 17 mCi of iodine-131 were released during refuelling and maintenance of the plant. These releases resulted in a maximum exposure in the surroundings of 0.7 mrem/yr. When it is considered that Obrigheim is one of the first PWR plants with zirconium-clad fuel rods and boric acid reactivity control, and when these release rates are compared with those of other PWR's equipped with zirconium-clad fuel, the release rates of Obrigheim are relatively low. In all newer PWR plants built by Siemens, considerable effort is made to reduce the radioactivity release rates. Continuous degasification by the pressurizer is provided to reduce noble gas release, which at Obrigheim turned out to cause about 70% of the gaseous release, by a factor of 10. Thermal degasification by the pressurizer, first applied during the Obrigheim refuelling, is very effective and reduced the specific activity by a factor of 500 during 100 hours of operation in the shut-down condition. Furthermore, gas decay tanks have been replaced by active charcoal delay beds, providing 50 days' delay time for xenon. Finally, the bleed boric acid will be completely evaporated, the water recirculated, and the off-gas sent to the delay beds. Through these measures, the newer PWR's of 600 to 1300 MW(e) will release less than 2000 Ci/yr of noble gases and 30 mCi/yr of iodine-131. These release rates cause a maximum integrated dose rate at ground level of less than 0.5 mrem/yr and can be tolerated at all sites.

2.2. Liquid releases

In German light-water reactor plants, the liquid waste treatment system invariably incorporates an evaporator system in addition to the

TABLE I. ACTIVITY RELEASE TO RIVER

Plant (River)	Year	Total release except ^3H (Ci/yr)	^3H release (Ci/yr)	Resulting dose if river is used as drinking water (mrem/yr)
250-MW(e) BWR Gundremmingen (Danube)	1967	3.5	-	3.5
	1968	2.6	14	2.5
	1969	1.6	17	1.5
	1970	1.6	30	1.5
250-MW(e) BWR Lingen (Ems)	1968	0.6	7.5	2
	1969	0.65	12	2
	1970	0.75	22	2
320-MW(e) PWR Obrigheim (Neckar)	1968	0.5	-	0.7
	1969	10.4	300	13 (1.4×10^{-2} ^3H -dose)
	1970	3.2	430	4 (1.8×10^{-2} ^3H -dose)

usual filters and ion exchangers. All radioactive water may thus be fed back into the plant, very slightly radioactive water, such as washing water, being added in controlled batches to the cooling water effluent. Table I, which lists the activity releases to the various rivers, may serve to demonstrate the liquid waste system capacities of the three large demonstration plants. The listed values of the mean river activity are defined as the exposure of a population drinking exclusively the river water homogeneously mixed with the radioactive effluents. Table I shows that the radioactivity release to the water, even if the water is used for drinking purposes, represents only 1 to 2% of the 150 mrem/yr value that has been recommended by ICRP.

In summary, it may be stated that the gaseous and liquid releases from operating light-water reactors in the Federal Republic kept maximum exposures of the population in the plant surroundings down to about 1% of natural activity exposures. Improved off-gas systems incorporated in plants currently under construction will certainly maintain — if not further reduce — this extremely low level; where releases under normal operating conditions are concerned, plant designs fulfil all requirements for urban siting.

3. SAFETY DESIGN FOR PRIMARY SYSTEM RUPTURE ACCIDENTS

As demonstrated in the preceding section, German light-water reactors have been designed to very stringent safety requirements for activity

releases under normal operating conditions. The same is true for accident prevention in general; the precautions against the most dangerous hypothetical accident, primary system rupture, are described below. Such precautions fall into three main categories: (a) design, manufacture, testing, and in-service inspection of primary system components, (b) core emergency cooling systems, and (c) safety containment.

3.1. Primary system safety

Where design, manufacture and inspection of primary system components, especially for the reactor vessel, are concerned, very strict requirements are observed in all countries. Specifications established by AEG and Siemens together with the TÜV generally follow international standards such as the ASME-Code, section III. However, our specifications contain requirements which are considerably more stringent than those of ASME-III; this can be demonstrated for the area of most concern, the quality of welds in reactor pressure vessels.

Repeated ultrasonic inspection of all welds after manufacturing and in service was required for the vessels of the demonstration reactors now operating, whereas it was only recently that this technique was required by reactor manufacturers in other countries; ASME-III still leaves the choice between radiographs and ultrasonics for most welds. The three most relevant features of ultrasonic weld inspection are described below.

(a) Extent of ultrasonic inspection

ASME requires inspection in seven directions per weld, while AEG and Siemens specifications call for 18 directions for submerged arc welds and 26 for electroslag welding. This large number of directions considerably reduces the probability of overlooking defects. Moreover, every inspection has to be performed by three inspectors independently, namely, by the vessel manufacturer, the reactor manufacturer and the TÜV.

(b) Size of tolerable defects

According to ASME, defects are tolerable if, in testing, they do not exceed the echo reflected from a 3/4-in. (19 mm) cylindrical hole in a test piece. German specifications set the standard against an 8-mm test bore.

(c) Accumulation of defects

While ASME-III does not distinguish between single defects and an accumulation of defects, Federal German specifications require repair of defects down to the equivalent of a 3-mm test bore if there are accumulations of more than six defects per 10-cm weld length or more than 20 per 100 cm.

To meet such stringent requirements, many difficulties had to be overcome by the pressure-vessel manufacturers. However, no incident has occurred to cause any important repair or other delay after installation

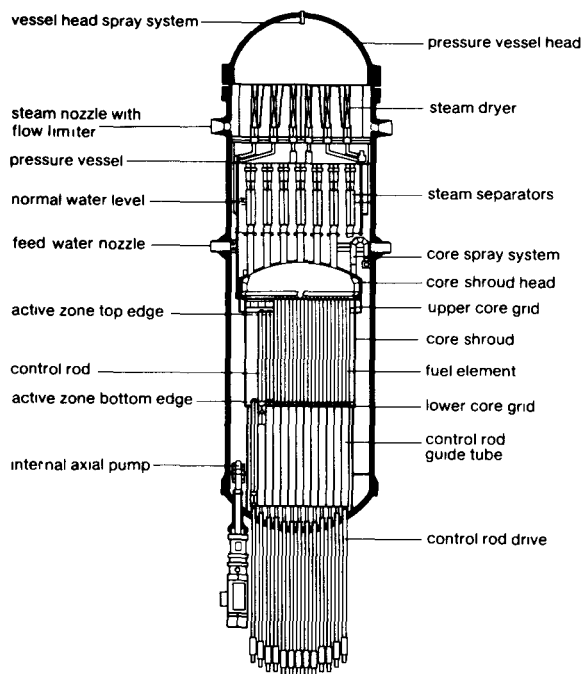


FIG. 2. AEG BWR, pressure vessel with internals.

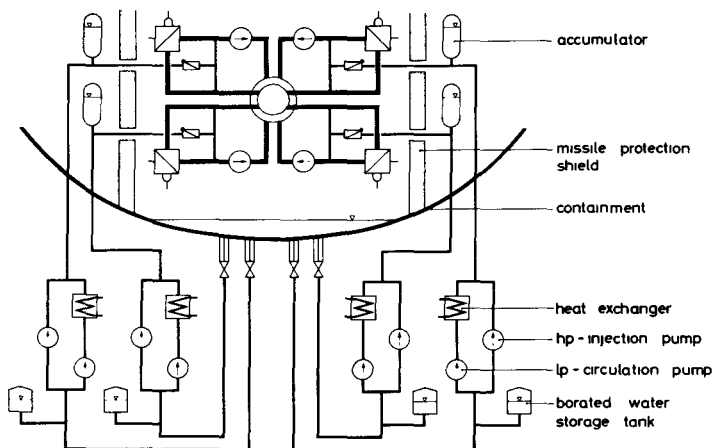


FIG. 3. Siemens PWR, flow scheme of emergency core-cooling system.

of any of the seven light-water reactor vessels now in operation or under preoperational testing in the Federal Republic.

3.2. Emergency core cooling

The basic design of the emergency core-cooling equipment for both Siemens PWR's and AEG BWR's follows the practice established in the United States. However, in this area as well, German safety designs have additional features in the newer commercial stations.

In the AEG BWR's, a first basic difference from other BWR's is the use of internal coolant recirculation pumps, as shown in Fig. 2. This does away completely with the big recirculation loops typical of all other light-water reactors of either the BWR or PWR type. It has the effect of decreasing the likelihood of the largest pipe break which serves as the basis for the maximum credible accident; this is now a break in one of the four steam pipes, and, since flow limiters are built into the steam nozzles of the pressure vessel, the diameter (about 300 mm) of these devices determines the maximum break area. Moreover, with no major pipes connected to the vessel below the core upper boundary, flooding of the core is lessened.

By taking advantage of the small pipe areas and the resulting long blowdown times in the maximum credible accident, it was economically feasible to design emergency core-cooling systems with a capacity such that the core would remain completely flooded under all hypothetical accident conditions.

For example, with a steam line rupture, it takes about 150 seconds for the reactor pressure to drop from 70 to 15 atm, at which level low pressure injection and core spray systems start to fill the vessel. By then, the water level outside the core shrouding has dropped considerably, but, because of the higher steam content, the whole space within the core shrouding is still filled with a steam-water mixture up to the steam separator exit. The core spray pump or any two of the four low-pressure coolant injection pumps would then suffice to maintain this water level, so that the fuel elements remain well cooled with very large redundancy during the whole accident. Similar redundancies keep the core flooded after a feedwater line rupture, which is the only other large hypothetical accident, as well as during blowdown occurring after breaks of any of the smaller pipes.

Thus, for the AEG BWR, core emergency cooling not only prevents the fuel from melting — which is the usual requirement — but keeps fuel cladding virtually at normal operating temperatures thus preventing even cladding ruptures; as a consequence, with pipe break accidents up to the maximum credible accident, virtually the only radioactive products released to the containment are those contained in the primary system water.

Finally, as a back-up, it is possible to flood the containment up to the level of the feedwater nozzles in the reactor pressure vessel.

The emergency core-cooling system of the Siemens PWR has an extraordinarily high degree of reliability. As the flow scheme in Fig. 3 shows, it is made up of four identical and independent systems, each connected to both the hot and cold legs of a loop, thus providing water injection from below the core and from the top. Except for the last check valves, the systems are well protected by a massive concrete missile-protection shield. All active components are arranged outside the

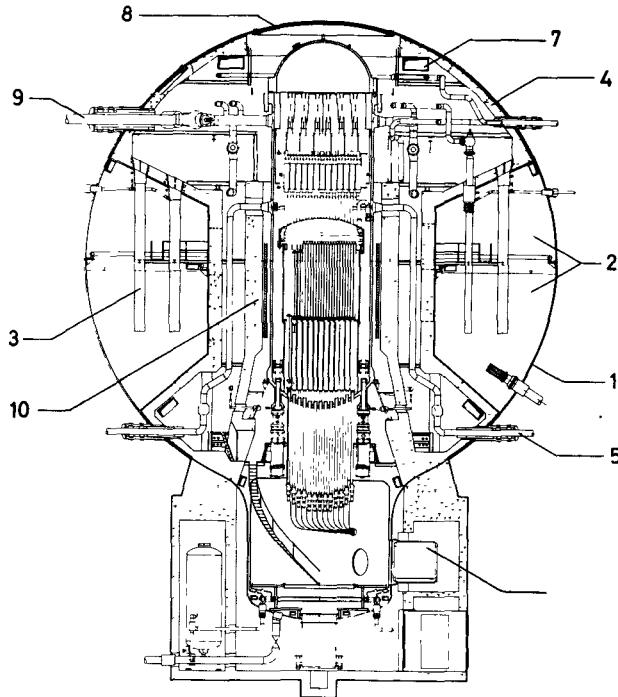


FIG. 4. AEG BWR, containment vessel: (1) primary and secondary containment shell; (2) condensation chamber; (3) condensation tubes; (4) missile protection concrete; (5) feedwater line; (6) personnel lock; (7) ventilation; (8) refuelling closure; (9) steam line; (10) biological shield.

containment sphere to protect them from all possible consequences of a leakage accident. Each of the four systems is equipped with an accumulator, a high-pressure injection pump, a low-pressure circulation pump, a decay-heat-removal exchanger, and a borated water storage tank. Each of the systems has its own emergency power diesel set and is connected to its own cooling system. The operation of any two of the four systems, including one injecting into the broken loop, keeps the maximum cladding temperature of the core below 800°C (1470°F) in the event of any leak of the primary system, so that dangerous ballooning of the fuel rods or a metal-water reaction is excluded. The redundancy of $4 \times 50\%$ of the emergency core-cooling system leads to a probability of less than 10^{-4} that its superior efficiency could not be fully guaranteed if a loss-of-coolant accident occurred.

3.3. Containment

Figure 4 shows the AEG containment design, which is a steel sphere of about 27-m diameter, designed as a pressure suppression system. The condensation chamber is situated within the sphere, concentric to the reactor. The steel walls of the condensation chamber and the sphere are completely protected against missiles and jet forces by self-supporting

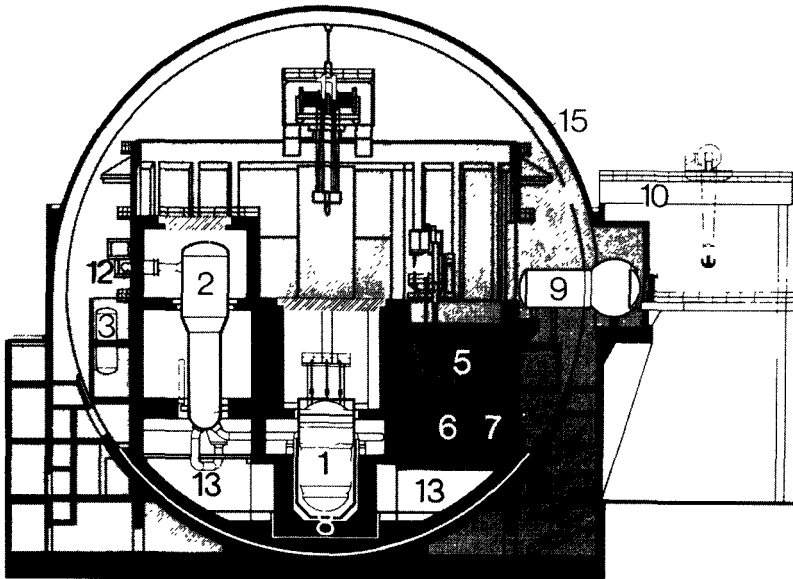


FIG. 5. Siemens PWR, reactor building and containment: (1) reactor pressure vessel; (2) steam generator; (3) accumulator; (4) fuelling machine; (5) fuel pit; (6) fuel rack; (7) fresh fuel transfer; (8) biological shield; (9) material lock; (10) transfer crane; (11) crane; (12) live steam pipe; (13) sump; (14) steel containment sphere; (15) secondary concrete shield.

concrete, which at the same time provides reliable support for the piping. These structures are surrounded by the reactor building, a solid reinforced-concrete structure.

The sphere is of a double-wall design, the inner wall supporting the accident pressure, the thinner outer wall providing a gap which is constantly kept at a pressure slightly below atmospheric. This gap can be re-vented into the containment after an accident for a minimum of 100 days until the design pressure of the containment is again reached. Thus, for at least 100 days, there is no activity release whatsoever after a major accident of primary system rupture; after 100 days, activity has decreased so that controlled gaseous release will be completely harmless.

It ought to be remembered that this no-release containment is provided in spite of the fact that the emergency core-cooling system of the AEG BWR prevents any release of activity from the fuel elements, as explained in the preceding section. Such over-redundancy is provided only to cope with urban siting problems.

The containment principle of the Siemens PWR, shown in Fig. 5, is unique in that it includes the spent-fuel storage pit, arranged directly adjacent to the reactor pit. This short distance, together with the ability of the fuelling machine to handle not only fuel elements, but also control rods, poison rods, source rods, etc., allows for an extremely short refuelling time of less than ten days from full power to full power. No handling of spent fuel takes place outside the closed containment. By means of continuous venting, the containment is kept fully accessible

during operation of the reactor for loading the shipping cask and for inspection and maintenance of the plant. The respective isolation valves are highly redundant and operated independently of each other. Non-welded penetrations of the containment shell are re-vented. All inside structures are not only designed to withstand reaction forces and pressure differentials for protecting the containment against all possible consequences of a leakage accident, but also to protect the primary system against influences from outside. The steel containment sphere is surrounded by a secondary concrete shield serving as extra protection against explosive shock waves or missiles. The space between the containment and the shield is separately vented to the stack via filters, thus forming a double containment. During pressure testing of the containments so far, it could be shown that the leak rate in all cases was less than 0.2 vol. % / day at design pressure.

It can be seen from these examples that both the BWR and PWR containment principles represent an extraordinary degree of protection against reactor accidents and external influence.

4. FURTHER DEVELOPMENTAL EFFORTS

It has been shown that modern light-water reactors in the Federal Republic of Germany are designed on the basis of safety standards exceeding those internationally used. Many of the safety features described are actually considered to be unnecessary exaggerations by both the reactor manufacturers and the German experts in reactor safeguards; however, they have been incorporated into standard designs in view of the psychological problems associated with urban siting in the Federal Republic. Co-operation between industry and safeguards authorities had been good while the standards were being developed. It was therefore very disappointing to the reactor industry when the Federal German Ministry of Education and Science announced in 1970 that the first urban reactor project — a process steam producing, dual-purpose plant for the chemical industry in Mannheim/Ludwigshafen — could not be licensed at that time. Instead, an increased developmental effort in reactor safety, funded by the Government, was announced to provide the necessary backing for urban siting.

This effort will certainly bring about many improvements in safe design, construction and operation of plants and their safety systems. Only reluctantly, however, did the reactor manufacturers Siemens and AEG accept to develop and test under this program further systems ensuring containment integrity, even for the extremely unlikely case of a sudden and complete rupture of the reactor pressure vessel. After a thorough investigation early in 1970, the Reactor Safety Commission (RSK) concluded that a major vessel rupture is of such remote probability that it need not be considered in the safety evaluation of the current light-water reactor plants. This evaluation, of course, took into account the very stringent specifications for reactor vessel manufacture and testing, as previously described in this paper. While — reluctantly — continuing with the development of safeguards against pressure vessel rupture, the industry still hopes to be finally successful in demonstrating that such safeguards are pointless. Apart from this, funds for the program will be used to

intensify what industry considers to be useful research and development for reactor safety. Continuously throughout the past, work in this field has been undertaken by Siemens and AEG, mostly in co-operation with the Federal Ministry of Education and Science and other institutions. Of the broad range of activities, the following few examples may characterize the research program on reactor safety which is currently being implemented or is in an advanced stage of planning:

In further investigations on heat-transfer conditions in the core during and after leakage accidents, AEG and Siemens are performing heat-transfer experiments on electrically heated rod bundles. These experiments are being carried out in the high-pressure region, i. e. under blowdown conditions, as well as in the low-pressure region, i. e. under the conditions of refilling the core. Experimental results will improve calculational methods and reduce pessimistic assumptions in the respective accident analyses. These tests will be done on bundles up to full scale.

The blowdown behaviour of the reactor vessel and the primary circuit and containment response are under investigation at the Battelle Institute, Frankfurt. The aim of these rather extensive experiments — the vessel employed is 11 metres high — is to investigate steam-water separation during two-phase blowdown, water plugging in steam tubes, blowdown rates, jet forces, and the spread of two-phase beams.

In addition to investigations on fracture-mechanics, special attention is being paid to improvement of ultrasonic testing procedures. In co-operation with the heavy steel industry, special remotely controlled testing methods are being developed for ultrasonic in-service inspection of reactor pressure vessels both from the interior and from outside. Recently, a method was successfully applied for taking the ultrasonic fingerprint of the pressure vessel of the 660 MW(e) Stade PWR plant.

An ambitious program should be mentioned which is still in the planning stage: the investigation of the behaviour of a molten core by theory and experiment. It is aimed at a better understanding of the associated phenomena, since at present — not only in the Federal Republic — the gap in the knowledge of these phenomena is covered by maximum pessimistic assumptions. It is hoped that, as a result of this program, more realistic assumptions on molten-core behaviour may be established, which will allow over-redundancies in the emergency core-cooling design to be abolished.

Finally, much attention is being paid to a thorough safety-system analysis of the current light-water reactors in the Federal Republic to harmonize the many safety systems, i. e. to detect weaker areas as well as to reduce overly redundant designs in other areas.

The safety record of the four operating light-water reactors in the Federal Republic of Germany is excellent, indicating an adequate safety design. As has been pointed out, for the generation of commercial light-water reactors now under construction in the Federal Republic, the safety standards are considerably more stringent than for the reactors already in operation; this stems from the necessity for urban siting. Industry does not expect further increases in number and redundancy of safety systems from further safety research, but the better harmonization of these systems.

RECENT DEVELOPMENTS IN THE UNITED STATES LIQUID-METAL FAST BREEDER REACTOR SAFETY PROGRAM

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Abstract-Résumé-Аннотация-Resumen

RECENT DEVELOPMENTS IN THE UNITED STATES LIQUID-METAL FAST BREEDER REACTOR SAFETY PROGRAM.

Safety development for the liquid-metal fast breeder reactor in the USA is based upon the principle of 'defense in depth'. A primary requisite of this approach is that the design, construction, and operation of reactor plants be in accordance with high quality standards and well understood technical bases, thus assuring the highest possible reliability and safety. Safety features not needed for normal operation are provided to accommodate mechanistically describable failures and malfunctions that cannot be completely precluded by design. Finally, consequence limiting features are provided, such as containment, to protect against events that cannot be characterized as absolutely impossible, even though they are of such remote probability that credible causative mechanisms cannot be defined. The major US development emphasis is on technology related to off normal operating conditions that have previously been postulated as potentially leading to major accidents. Less emphasis is placed on the accommodation of hypothetical consequences of undefined events. Recent progress leads to optimism that the safety of liquid-metal-cooled fast breeder reactors can be firmly established without undue economic penalties: the Doppler effect has been established theoretically and experimentally as a major prompt negative feedback mechanism for oxide-fuelled liquid-metal fast breeder reactors; analysis and experiment have provided a basis for understanding the effects of sodium voiding, Transient Reactor Test experiments have shown that the work energy availability from fuel-coolant interactions is much smaller than the thermodynamic maximum; significant sodium superheat is not expected under reactor operating conditions, gas escaping from fuel pins has been shown experimentally to be an unlikely source of fuel failure propagation; improved and experimentally verified accident analysis methods have been developed; modelling of the accident response of reactor internals, vessels, vessel covers, and support structures has been developed; theoretical aerosol models have been confirmed by engineering scale experiments, which should reduce present conservative approaches.

RECENTS DEVELOPPEMENTS DANS LE PROGRAMME DES ETATS-UNIS D'AMERIQUE DE SECURITE POUR LES REACTEURS SURGENERATEURS RAPIDES REFROIDIS AUX METAUX LIQUIDES.

Le développement de la sécurité pour les réacteurs surgénérateurs rapides refroidis aux métaux liquides aux Etats-Unis d'Amérique est basé sur le principe de la « défense en profondeur ». Une conséquence importante qui en découle est que le projet, la construction et le fonctionnement des réacteurs doivent s'appuyer sur des standards de haute qualité et des bases techniques bien comprises, assurant ainsi

le maximum de sûreté et de sécurité. Des dispositifs spéciaux de sécurité, non nécessaires pour une utilisation normale, sont fournis pour tenir compte des pannes et des mauvais fonctionnements mécaniques qui ne peuvent être totalement éliminés dans la conception du projet. En définitive des dispositifs tels que le confinement, destinés à limiter les conséquences de pannes éventuelles, sont installés afin d'assurer la protection contre des événements qui ne peuvent être considérés comme absolument impossibles, même si leur probabilité est si faible qu'il n'est pas possible de prévoir quelles pourraient en être les causes. Aux Etats-Unis l'accent est particulièrement mis sur la technologie concernant le fonctionnement dans des conditions anormales qui, à l'avance, ont été reconnues susceptibles de produire des accidents graves. On accorde moins d'importance aux moyens de résoudre les problèmes hypothétiques posés par des événements impossibles à prévoir. Des progrès récents permettent d'être optimistes quant aux possibilités d'assurer la sécurité des réacteurs surgénérateurs rapides refroidis aux métaux liquides sans augmentation exagérée du prix de revient: la théorie et l'expérience ont établi que l'effet Doppler était un important mécanisme de contre-réaction négative rapide pour ce type de réacteur, l'analyse et l'expérimentation ont permis de comprendre le mécanisme de la formation de lacunes dans le sodium, des tests effectués à l'aide du Transient Reactor Test (TREAT) ont démontré que l'énergie fournie par les échanges de chaleur entre le combustible et le fluide de refroidissement est beaucoup plus petite que le maximum thermodynamique, une surchauffe importante du sodium dans les conditions de fonctionnement du réacteur n'est pas attendue, il a été montré expérimentalement comme très improbable que les gaz s'échappant des aiguilles de combustible puissent être la source de propagation de failles dans celui-ci, des méthodes améliorées et expérimentalement vérifiées d'analyse des accidents ont été développées, la simulation de réponses aux accidents des organes internes du réacteur, des cuves, des enveloppes de protection et des structures de soutien a été étudiée, des modèles théoriques d'aérosol ont été confirmés par le comportement de maquettes expérimentales, ce qui permettrait de réduire à l'avenir les marges de sécurité appliquées aux systèmes actuels.

СОВРЕМЕННЫЕ РАЗРАБОТКИ ПО ПРОГРАММЕ БЕЗОПАСНОСТИ РЕАКТОРОВ -БРИДЕ -РОВ НА БЫСТРЫХ НЕЙТРОНАХ С ЖИДКОМЕТАЛЛИЧЕСКИМ ТЕПЛОНОСИТЕЛЕМ В США.

Разработка вопросов безопасности реакторов-размножителей на быстрых нейтронах с жидкометаллическим теплоносителем в США основывается на принципе более глубокого изучения проблемы. Основное требование такого подхода заключается в том, что проектирование, сооружение и эксплуатация реакторных установок ведется на основе использования высококачественных стандартов и хорошо изученной технической базы, обеспечивающих максимально возможную надежность и безопасность. Характеристики безопасности, в которых нет необходимости при нормальной эксплуатации, определяются для механически описанных аварий и случаев ненормальной работы, которые нельзя полностью предотвратить в процессе проектирования. Наконец, предусматриваются меры ограничения возможных последствий, такие, как создание защитной оболочки для предотвращения случаев, которые не могут быть характеризованы как абсолютно невозможные, даже если они настолько маловероятны, что возможные причины их возникновения не могут быть определены. В США основные разработки вопросов безопасности касаются технологии, связанной с условиями ненормальной эксплуатации. Имеются в виду условия, которые могут привести к крупным авариям. Меньшее внимание уделяется исследованию гипотетических последствий неопределенных аварийных ситуаций. Последние достижения приводят к оптимистическому выводу о том, что безопасность реакторов-размножителей на быстрых нейтронах с жидкометаллическим теплоносителем может быть надежно обеспечена без дополнительных экономических затрат по следующим причинам. Теоретически и экспериментально установлено, что эффект Доплера является основным механизмом мгновенной отрицательной обратной связи для реакторов-размножителей на быстрых нейтронах с жидкометаллическим теплоносителем, работающих на окисном топливе. Анализ и эксперименты обеспечили основу для понимания роли пустот в натрии; эксперименты по исследованию переходных процессов в реакторе показали, что рабочая энергонапряженность топлива, определяемая взаимодействием с теплоносителем, значительно ниже термодинамического максимума; в условиях эксплуатации реактора не ожидается существенного перегрева натрия; экспериментально показано, что выделение газа из топливных элементов не приводит к распространению повреждений топливных элементов; разработаны усовершенствованные и экспериментально опробованные методы анализа аварий; разработаны способы моделирования поведения при авариях внутренних узлов реактора, корпусов, внешних оболочек и опорных конструкций; теоретические модели аэрозолей подтверждены масштабными инженерными экспериментами, которые должны исправить современный консервативный подход.

PROGRESOS RECIENTES EN EL PROGRAMA DE LOS ESTADOS UNIDOS DE AMERICA EN MATERIA DE SEGURIDAD DE LOS REACTORES REPRODUCTORES RAPIDOS DE METAL LIQUIDO.

En los Estados Unidos, los trabajos relativos a la seguridad de los reactores reproductores rápidos de metal líquido se basan en el principio de «defensa escalonada». Para ello es condición fundamental que el proyecto, construcción y funcionamiento de los reactores se ajusten a normas de alto nivel y bases técnicas bien conocidas, garantizando de este modo la mayor regularidad funcional y seguridad posibles. Se adoptan precauciones de seguridad, no necesarias en circunstancias normales, en prevención de averías y fallos de funcionamiento de origen mecánico, imposibles de eliminar completamente por las condiciones de proyecto. Finalmente, se prevén también medios para atenuar las consecuencias, tales como la contención, como protección contra eventualidades que no pueden juzgarse absolutamente imposibles, si bien su probabilidad es tan remota que no es dado precisar los fenómenos causales verosímiles. En los Estados Unidos se presta especial atención a la tecnología relacionada con las condiciones anormales de funcionamiento que se consideran causas potenciales de accidentes importantes. Se atribuye menos importancia a la prevención de las consecuencias hipotéticas de sucesos no claramente definidos. Los progresos recientes inducen a pensar que será posible establecer sobre una base sólida la seguridad de los reactores reproductores rápidos refrigerados por metal líquido, sin cargas económicas excesivas. Se ha demostrado teórica y experimentalmente que el efecto Doppler es un medio importante de realimentación negativa inmediata en los reactores reproductores rápidos de metal líquido y combustible en forma de óxido; los análisis y experimentos efectuados sirven de base para conocer los efectos de la formación de vacíos en el sodio, los experimentos realizados con el Transient Reactor Test (reactor para el estudio de perturbaciones transitorias) han demostrado que la disponibilidad de energía de trabajo debida a las interacciones entre el combustible y el refrigerante es mucho menor que el máximo termodinámico; no se espera un sobrecalentamiento importante del sodio en las condiciones de funcionamiento del reactor; se ha comprobado experimentalmente que es improbable que los escapes de gases de las agujas combustibles originen la propagación de roturas en el combustible, se han elaborado métodos perfeccionados y verificados experimentalmente de análisis de accidentes; se han preparado modelos para predecir el comportamiento en caso de accidente de los dispositivos internos, vasijas, tapas de vasijas e infraestructuras de reactor, se han comprobado, mediante experimentos a escala industrial, modelos técnicos de aerosoles, lo cual debería atenuar la rigurosidad de los criterios hasta ahora aplicados.

INTRODUCTION

Liquid Metal Fast Breeder Reactor (LMFBR) safety development in the United States is based upon the principle of defense in depth. A primary requisite of this approach is that the design, construction, and operation of reactor plants be in accordance with high quality standards and well understood technical bases, thereby assuring the highest possible reliability and safety.

A review of reactor operating experience indicates that a large fraction of malfunctions, including those that are safety related, could have been prevented by straightforward quality assurance practices, particularly through additional attention to design details, quality control during fabrication and construction or proper operation. Appropriate operating procedures and comprehensive specifications and standards for reactor equipment provide the basis for reactor owners, designers, contractors and suppliers to assure that reliability and safety objectives will be met. It is only through this disciplined engineering approach that the requisite high degree of reliability and safety can be attained and that the technical difficulties and uncertainties associated with the phenomenology and accommodation of accidents and failures can be avoided or minimized.

To provide a defense in depth, engineering safety features and reactor shutdown systems not needed for normal operation are included to arrest and safely accommodate mechanistically describable failures and malfunctions which cannot be completely precluded by design. To establish a basis for design of such features, analytical and experimental

information has been developed concerning primary system failures, sodium fires, fuel element failure propagation, passage of large gas bubbles through the core, and other occurrences in which a single malfunction could lead to significant safety consequences.

Finally, consequence limiting features, such as containment, are provided to protect against events which cannot be characterized as absolutely impossible, even though they are of such remote probability that credible causative mechanisms cannot be defined. Hypothetical events are used to evaluate the design adequacy of such features, and margins to accommodate large energy releases are determined and optimized. Effort is presently focused on hypothetical events such as ramp addition of reactivity or loss of coolant flow which are postulated to occur concurrently with failure of the reactor shutdown system. Doppler effect, timing and location of subsequent failures, rates and pattern of voiding of the core, molten fuel-sodium interactions, and the cooling of any core debris are significant factors in these accident evaluations.

The major United States safety development emphasis is on technology related to off-normal operating conditions which have previously been postulated as potentially leading to major accidents. Emphasis is also placed on the accommodation of hypothetical consequences of undefined events. The rationale for this approach and the objectives, scope, priorities and interrelationships of the elements comprising this program are described in the LMFBR Program Plan, Volume 10, Safety, WASH 1110.

The major part of the United States LMFBR Safety Program is conducted in direct support of the 400 MW(th) Fast Test Reactor presently under construction at Hanford. The work is so paced and the characteristics of the Fast Test Reactor have been so chosen that the information developed in the program is of direct and immediate use in the USAEC's LMFBR Demonstration Plant Program.

Further details of recent progress and current efforts and plans are presented below, including the locations where the work in this national program is being performed.

INCIPIENT SODIUM SUPERHEAT AND SODIUM VOIDING

Initial work [1] on incipient superheat included Argonne National Laboratory tests [2] which showed that inert gas concentration in surface cavities can significantly reduce incipient boiling superheat in static systems. Further work at Atomics International [3] and Brookhaven National Laboratory [4] in flowing systems show a definite decrease of superheat with increasing flow rate. These results suggest that incipient superheat will not play a major role in the behavior of LMFBR's during hypothetical accidents. While a theoretical explanation of low superheats is not fully developed [5], factors may include wall pressure fluctuations associated with turbulent flow [3,4,6], the presence of minute inert gas bubbles in the coolant [7], and the presence of a relatively high partial pressure of inert cover gas present in the surface cavities of the boiling surface [8].

Sodium expulsion which is postulated to result from different hypothetical accident sequences (such as loss-of-flow and overpower transients) may occur as a result of coolant boiling, fission-gas release, or molten fuel-coolant interaction [11]. Out-of-pile and in-pile experiments are underway at Atomics International, General Electric and Argonne National

Laboratory. For liquid-metal transient boiling under typical LMFBR geometries, pressures, and heat fluxes, the slug-annular flow regime appears to prevail [9]. The validity of a single-bubble model [10] has been illustrated by water and freon tests which simulated voiding due to fission-gas release in pin bundles. Tests by General Electric are planned to assess voiding rates resulting from fission gas release from failed fuel.

MOLTEN FUEL-COOLANT INTERACTIONS

A variety of mechanistic models for fuel-coolant interaction have been proposed by a number of investigators [14-17]. Current models are primarily concerned with available work by the expanding coolant vapor. Acoustic and inertial constraint, mixing times and possible heat transfer cutoff mechanisms are important considerations.

Initial conditions which appear to affect the nature and extent of fuel-coolant interactions are distribution and state of the fuel and coolant, quantity and disposition of gas and coolant vapor, the degree of fuel fragmentation, rate of mixing, and configuration of the system. Although a theoretical thermodynamic efficiency of the conversion of thermal energy to mechanical work as large as 30% is calculated by Hicks and Menzies [13], all experimental evidence to date for UO_2 -sodium systems show conversion ratios lower by 2 to 4 orders of magnitude. However, it should be noted that not all situations of interest have been examined, e.g., the consequences of sodium reentry.

Experiments to date have included a series in the TREAT reactor performed by Argonne National Laboratory to determine the pressures and mechanical energies that result from the interaction of molten fuel and sodium. These tests utilized multi-pin arrays of unirradiated UO_2 clad with stainless steel, in stagnant sodium in an autoclave [18]. The autoclave contained a movable piston which provided an inertial load corresponding to about a 1-meter head of sodium. The results of four tests shown in TABLE I indicate that the presence of bond and fission gas tend

TABLE I. Results of Piston Autoclave Tests in TREAT

Test No.	S3	S4	S5	S6
No. of fuel pins	5	5	5	7
Bond gas	He-Bonded	He-Bonded	Evacuated	He-Bonded
Fuel mass, g	205	205	205	286
Fuel to sodium area ratio (in fuel array)	0.45	0.45	0.45	1.06
Initial sodium temperature, °C	165	165	170	210
Fission energy, cal/g- UO_2 ^a	460	720	509	493
Reactor period, msec	53	34	46	38
Fission energy at first pressure pulse, cal/g- UO_2	440	480	430	380
Amplitude largest pressure pulse, atm	37	122	200	140
Calculated conversion from nuclear to mechanical energy	2.1×10^{-5}	2.5×10^{-4}	1.9×10^{-3}	1.5×10^{-3}

^aSpecific energy of UO_2 at the liquidus point is 335 cal/g and vaporization occurs at 376 cal/g at atmospheric pressure. Energy deposition in fuel occurred in about 100 msec.

to mitigate the molten fuel-coolant interaction, presumably by inhibiting heat transfer and by providing compliance to the system. In no case was there evidence of a massive, coherent interaction of molten fuel and coolant leading to a violent sodium-vapor explosion. Instead, a number of sharp pressure pulses were observed in the hundred-atmosphere range which seemed to correlate with the failure of individual pins. The calculated conversion ratio of thermal to mechanical energy was remarkably small, ranging from 2×10^{-3} to 2×10^{-2} . These results are similar to the TREAT tests performed in flowing sodium in the MARK II loop.

In related tests at Atomics International [19], stainless-steel-clad UO_2 was melted by electrical resistance heating under a 6 meter column of sodium at 650°C . The initial experiment deposited 960 cal/g in the UO_2 in a 250 msec period and gave a measured conversion ratio of thermal to mechanical energy of about 3.5×10^{-3} -- the same range as the results in TREAT.

FUEL-FAILURE PROPAGATION

The potential for propagation of failures within a fuel assembly and between assemblies are areas of significant work in the United States.

In spite of stringent quality control efforts, failure of some pins in a reactor core due to random clad rupture, error in fuel enrichment or local blockage may occur [20]. The potential for propagating failure due to the release of fission gas is being investigated in out-of-pile tests at Argonne National Laboratory [10]. It has been concluded from analysis that a rapid release of gas would be of sufficiently short duration that coolant flow would be reestablished prior to overheating of adjacent pins. Experimental investigations of the effects of a slower flowing persistent gas jet indicate that the surface temperature increases moderately at the point of jet impingement, but not to an extent which would suggest failure propagation is a concern.

Work is underway at General Electric in which in-pile tests are used to assess potential blockage due to pin swelling caused by chemical interaction between sodium and the fuel of a failed pin. In out-of-pile work, Oak Ridge National Laboratory is investigating the effects of local blockages within an assembly using 19 electrically-heated elements in a sodium loop.

A series of larger scale in-pile tests are planned to study pin-to-pin failure. Initial interest is focused upon confirmation of out-of-pile results of gas release, the behavior of the release of small amounts of molten fuel, and the consequences of a loss-of-flow accident. A conceptual design of forced circulation package loops for use in the Engineering Test Reactor is complete. These 1.2 MW loops, scheduled for operation in 1973, are sized for a triangular array of 37 pins, 7 feet long.

Initial work involving assembly-to-assembly propagation has been of a theoretical nature [21-22]. This work considers hypothetical incidents in which significant failure propagation within the assembly has been assumed to have occurred, or where a whole assembly has been blocked. The postulated failure due to high pressure pulse generation from credible molten fuel-coolant interactions will be assessed, as will lesser important mechanisms such as thermal melt-through.

ACCIDENT ANALYSIS METHODS

Marked progress in the capability to analyze fast reactor accidents has led to a significant reduction in the degree of conservatism previously employed to compensate for uncertainties.

The MELT-II code [23] is an initiating accident-analysis code whose principal merits are speed, small computer memory requirements, and breadth of areas treated. FREAMD-1, another accident initiating code [25], operates at a similar level of sophistication. One of its features is the capability to analyze assembly-to-assembly propagation employing user-selected criteria as triggers.

SAS2A [26] is a multichannel code derived from the SASIA code. The mechanical behavior of fuel and cladding is treated in detail in the code with irradiation effects included in the analysis. SAS2A computes voiding caused by sodium boiling, the escape of fission-product gas upon clad rupture, and by molten fuel coolant interaction. Sodium-voiding analysis due to boiling using the many-bubble slug-ejection model of SAS2A [12] is shown in Figure 1. Calculated voiding patterns during a pump coastdown accident illustrate the expected effect of incipient superheat on the rate of voiding. As can be seen from this figure an initial vapor bubble is formed near the top of the core after the coolant temperature has exceeded its boiling point by the assumed value of superheat. The initial momentum of the coolant carries the first bubble upward into the cooler reflector, and since the coolant is continuously being heated a second bubble is formed below the first one, creating an intermediate liquid slug between the two vapor bubbles. This process continues for about 250 milliseconds after which the vapor begins to expand in both directions in the channel. At this point the friction loss due to sodium vapor flow exceeds the head of the coolant, resulting in dryout of the channel. Additional clarifying experimental and analytical work is underway in this area.

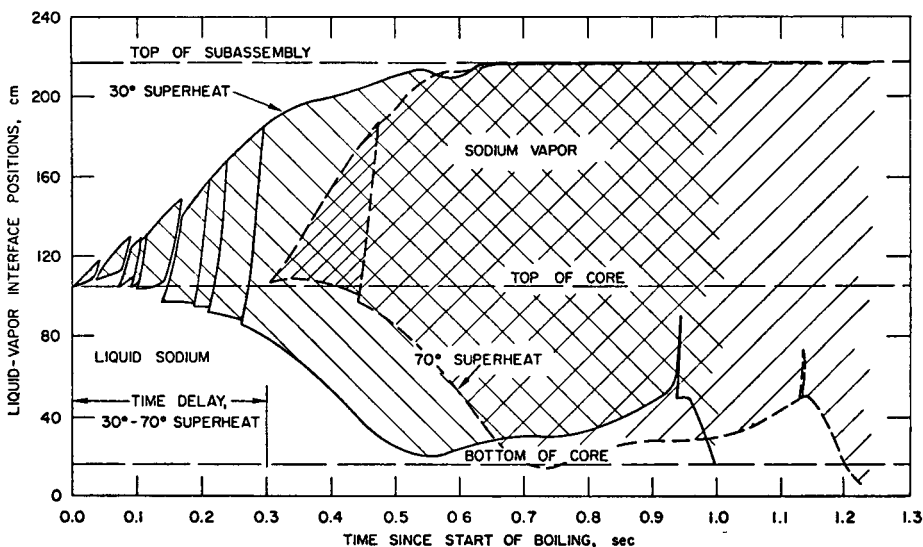


FIG. 1. Sodium void expulsion as calculated by SAS2A code for 30 and 70°C superheats, and for coastdown accident conditions.

If the accident being calculated by the MELT-II, FREAMD-1 or SAS2A progresses toward disassembly, the calculation can be coupled to the disassembly code VENUS [24], which is a two-dimensional hydrodynamics disassembly code developed at the Argonne National Laboratory.

The VENUS code can handle "sodium-in" core regions as well as "sodium-out" regions, which is especially important for evaluation of flow coastdown accidents in larger reactors.

The FISFAX code [15] was developed by Atomics International to evaluate the accident dynamics which follow the rupture of oxide fuel in a power ramp, leading to rapid expulsion of sodium and fuel. This code takes into account the negative feedback effect of fuel leaving the core region following initial fuel failure.

MECHANICAL-RESPONSE ANALYSIS OF REACTOR STRUCTURES

Improvements in two-dimensional axisymmetric calculational codes for predicting the effects of shock and blast loadings on reactor structures, including internals and primary containment boundary, have been accomplished. REXCO-H [27], which treats all materials inside the reactor vessel as fluids during the initial part of the accident when pressures are high, describes the effects of the impact of the sodium slug upon the vessel, and subsequent response of the vessel cover and hold-down bolts as a function of time. This code, which was developed at Argonne National Laboratory, has predicted experimental results of model tests by the UKAEA, CEA-Euratom, and the U. S. Naval Ordnance Laboratory with good agreement, as shown in Figure 2, particularly in cases where the configuration was not overly complex [28].

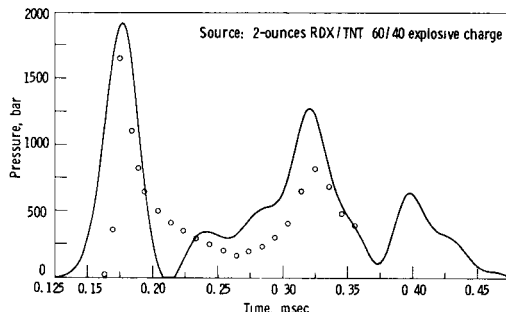


FIG. 2. Comparison of REXCO-H computed pressures (curve) with measured values (circles) at gauge-3 for test with bar charge in water.

A new code called REXCO-I is being developed [29] which will take into account the shear strength of materials during the postburst portion of the accident. It will permit the evaluation of accidents of small magnitude for which the hydrodynamic code may provide nonconservative results.

DOPPLER EFFECT

The Doppler effect has been established theoretically [30] and experimentally as a major prompt negative reactivity feedback for current designs of oxide-fuelled LMFBR's. As a result of the efforts

noted below, uncertainty in calculation of the Doppler reactivity effect caused by fuel temperature changes in LMFBR's now appears to result largely from nuclear data uncertainties (estimated by one source as $\pm 15\%$) [31] and from calculational method approximations, rather than from inadequacies in the theory of Doppler-broadened cross-sections. Doppler reactivity uncertainties originating in reactor-neutronics methods [32-36] are shown in TABLE II. These data were obtained from plate-type samples in the ZPR-3 and ZPR-9. Deviations of calculated results from experimental measurements [32, 37-44] are also shown. A new facility, the Variable Temperature Rodded Zone (VTRZ) [45] is planned for ZPR-6, which will allow use of rod-geometry in a heated zone of the core region surrounding the Doppler sample.

TABLE II. Doppler Uncertainties

A. Modeling Effects		Computed Sensitivity	B. Experimental-Analytical Comparisons	
Data		$\pm 15\%$	Pu-U Experiments	Calculated Deviation From Measurements
Calculational Factors				
Resonance Overlap,			ZPR-3 Assy 53	$\pm 7\%^a$
Multigroup Structures,			ZPR-9 Assy 26	$\pm 3\%^a$
Perturbation Approximations,			ZPR-3 Assy 51 (Spatial	$\pm 40\%^b$
Multilevel Effects		$\pm 10\%$	Measurements)	$-25\%^b$
Determination of Resonance Spectrum		$< \pm 30\%$	All Uranium Cores	$-30\%^a$
Space Dependent Effects		$< \pm 10\%$		
Design And Operational Variables				
Fuel Burnup		$\pm 10\%$		
Control Rod Influence		Design Dependent		
Gross Temperature Variation		—		

^a Typical-about one-sigma - deviation

^b Total Deviation

In a related effort, the SEFOR reactor project [46] is performing full core heating experiments to determine the Doppler reactivity effect at fuel temperatures and in a neutron spectrum characteristic of current LMFBR designs [47]. The initial results of the zero power [48] and full power [49] tests show good agreement with calculation. The Doppler reactivity effect magnitude inferred from these SEFOR measurements is approximately 1% less than the predicted magnitude (assuming zero ^{239}Pu Doppler effect and a nominal value for fuel conductivity). The experimental uncertainty (one-sigma) is currently estimated to be $\pm 17\%$. Transient tests (sub-prompt and super-prompt critical) and tests in a harder-neutron spectrum core (240 keV median fission energy) are planned to be completed in 1972.

The current agreement between calculation and measurement for small-sample (critical assembly) measurements and for the SEFOR measurements (in which ^{239}Pu Doppler is a small contributor) is highly encouraging. It is hoped that with continued emphasis, an uncertainty band for the predicted Doppler effect for large LMFBR's of the order of $\pm 25\%$ will be achieved.

SODIUM BURNING AND AEROSOLS

Sodium burning and aerosol investigations are underway [50,51] to aid in the selection of leak detection devices, to develop fire and leakage control techniques and to investigate the temperatures, pressures and aerosol characteristics that might exist following major accidents.

Aerosol transport characteristics have been modelled by Atomics International [56] and Brookhaven National Laboratory [57] utilizing the hypothesis that the coagulation of aerosols depends on collisions resulting from Brownian motion of the suspended particles [53-55]. Mechanisms for particle removal including fallout, plating on the walls and leakage from the containment volume have been added to the model. For large concentrations, gravitational coagulation brought about by differential rates of settling becomes an important removal factor that has also been modelled [58].

The aerosol model shows that for any substantial aerosol concentrations, rapid removal occurs through joint processes of particle growth and fallout. Model predictions have been validated by comparison with experiments over a wide range of vessel volumes, including a 30-foot tall vessel at Atomics International. These studies show that the natural removal mechanisms that occur following accidents involving a sodium release are extremely effective in limiting the potential consequences. However, several second-order phenomena remain which merit additional consideration, such as re-suspension of deposited particles under conditions of severe turbulence, effect of electrical charge on aerosol agglomeration, collision efficiency of aerosols agglomerating due to differences in settling velocities, and settling velocities as a function of aerosol density and shape.

Theoretical and experimental programs on sodium pool and spray fires are being conducted. SOFIRE II [59] is a computer code which solves a set of theoretical equations describing the simultaneous process of heat and mass transfer during a single-cell sodium pool fire to predict burning rate, temperature and pressure. Agreement between theory and experiment has been good in cell volumes up to 2200 ft³ [52]. Another code, CACECO, developed at Hanford Engineering Development Laboratory, provides a more detailed pressure time history for sodium released into a confined volume. Future work will consider a fire in a cell which is open to the containment atmosphere.

Spray fires experiments are being conducted at Atomics International under varying conditions of humidity and oxygen concentration. Based on tests to date, it appears that pressures from spray-type fires will be substantially less than those associated with theoretical thermodynamic burning.

CONCLUSION

Results to date from the LMFBR safety program have been encouraging with respect to meeting the objectives outlined in the introduction. LMFBR's are capable of being built for reliable and safe operation under normal and near-normal operating conditions. Engineered safety features will be incorporated capable of precluding or accommodating unlikely failures and other events. Work on hypothetical core disassembly accidents permits more complete and detailed modelling resulting in more

realistic estimates of energy release and assessments of effects. Further work in each of these areas will be accomplished with the confident expectation of adequate safety assurance for each specific LMFBR plant.

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EXPERIMENTAL WORK ON FAST REACTOR SAFETY

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Abstract—Résumé—Аннотация—Resumen

EXPERIMENTAL WORK ON FAST REACTOR SAFETY.

Fuel-pin irradiations were performed to obtain information on the mechanisms and consequences of pin failures in the event of a disturbance in the sodium cooling. The experiments consisted of: performance tests at fuel rod powers of 650 W/cm and high burn-ups to evaluate the behaviour of fuel and canning; short-term irradiations of fuel pins with internal gas pressure and at high sodium temperature (parallel to these experiments a number of out-of-pile high temperature burst tests were performed); fuel-rod irradiations in specially designed capsules, in which the fuel rods were brought to failure by a sudden decrease in heat removal which was achieved by expulsion of NaK from an annulus around the fuel pin. In addition, experiments were performed to investigate the mode of sodium boiling. Instrumentation, for instance void detectors, were developed for sodium expulsion experiments.

The above-mentioned experiments concerned the class of minor accidents which, if undetected, may hypothetically lead to a major accident by which part of the core will be vaporized. This material will become airborne for longer periods. To evaluate such accidents, knowledge of the behaviour of aerosols in closed systems is required. Investigations on this behaviour were carried out during the past few years at Petten. Different materials were dispersed by the exploding wire technique. The aerosol formation and the concentration decay were studied. The parameters were, among others, vessel dimensions, temperature, and energy of formation.

TRAVAUX EXPERIMENTAUX SUR LA SECURITE DES REACTEURS A NEUTRONS RAPIDES.

Il a été procédé à des irradiations de crayons de combustible pour obtenir des informations sur les mécanismes et les conséquences d'une défaillance d'un crayon dans le cas d'une perturbation dans le système de refroidissement au sodium. Les expériences ont consisté en: des tests de performance à des puissances dissipées du crayon de combustible de 650 W/cm à des taux élevés de combustion pour évaluer le comportement du combustible et de l'enveloppe; des irradiations à court terme de crayons de combustible sous une pression interne de gaz et à une température élevée du sodium (parallèlement à ces expériences un certain nombre de tests de rupture ont été exécutés hors pile à haute température); des irradiations de crayons de combustible dans des capsules spécialement conçues, dans lesquelles le crayon de combustible est mis en défaillance par une décroissance soudaine de l'extraction de chaleur par expulsion du NaK hors d'un anneau entourant le crayon. En outre des expériences ont été exécutées pour étudier l'ébullition du sodium. Une instrumentation, par exemple des indicateurs de vide, a été développée pour les expériences concernant l'expulsion du sodium.

Ces expériences sont considérées comme correspondant à la classe des accidents mineurs qui, s'ils ne sont pas détectés, peuvent hypothétiquement conduire à des accidents graves à la suite desquels une partie du cœur se vaporisera. Les produits seront alors présents dans l'air pendant longtemps. Pour l'évaluation de ce type d'accidents, une connaissance du comportement des aérosols en système clos est requise. Des recherches sur ce comportement ont été effectuées ces dernières années à Petten. Différentes matières ont été répandues par la méthode du filament explosant. La formation de l'aérosol a été étudiée de même que l'affaiblissement de la concentration, les paramètres étant entre autres les dimensions de l'enceinte, la température et l'énergie de formation.

ЭКСПЕРИМЕНТАЛЬНЫЕ ИССЛЕДОВАНИЯ ПО БЕЗОПАСНОСТИ РЕАКТОРОВ НА БЫСТРЫХ НЕЙТРОНАХ.

Проведены радиационные испытания нескольких топливных элементов для получения данных о механизмах и последствиях разрушения топливных элементов в случае возмущения жидкометаллического теплоносителя. Проводились следующие эксперименты: испытания ра-

боты топливных элементов при линейной мощности 650 Вт/см и больших выгораниях для проверки поведения топлива и его оболочек; кратковременные радиационные испытания топливных элементов в условиях внутреннего давления газов и высокой температуры жидкого натрия (параллельно с этими экспериментами было проведено несколько вне реакторных высокотемпературных испытаний); радиационные испытания топливных элементов в специальных ампулах, в которых топливные элементы подвергались разрушению путем мгновенного уменьшения теплоотвода (последнее достигалось удалением эвтектики NaK из кольцевого зазора вокруг топливного стержня). Проведены дополнительные эксперименты по изучению механизма вскипания жидкого натрия. Для проведения экспериментов по удалению натрия были разработаны специальные приборы, например датчики для регистрации образования пустот.

Считается, что описанные выше эксперименты относятся к классу небольших аварий. Последние, будучи не обнаруженными, могут, в принципе, привести к более серьезной аварии, во время которой часть активной зоны испарится. При этом испарившийся материал будет в течение продолжительного периода времени находиться во взвешенном состоянии. Для оценки вероятности такого рода аварий необходимо знать поведение аэрозолей в замкнутых системах. Этот эффект изучался на протяжении последних лет в Петтен. Диспергирование материалов производилось методом взрывного распыления проволочек. При этом исследовалось образование аэрозолей и уменьшение их концентрации со временем. Наряду с другими параметрами, были изучены следующие: габариты корпуса реактора, температура и энергия, при которой образуются аэрозоли.

TRABAJOS EXPERIMENTALES SOBRE LA SEGURIDAD DE LOS REACTORES RAPIDOS.

Se han realizado irradiaciones de varillas de combustible para obtener información sobre el mecanismo y las consecuencias de la rotura de una varilla en caso de perturbación del sistema de refrigeración con sodio. Los experimentos consistieron en: pruebas de funcionamiento a potencias de la varilla de combustible de 650 W/cm y a altos grados de quemado, para evaluar el comportamiento del combustible y del envainado; irradiaciones a corto plazo de varillas de combustible con presión interna de gas y a una temperatura alta del sodio (paralelamente a estos experimentos se han ejecutado cierto número de pruebas de rotura fuera del reactor a alta temperatura); irradiaciones de varillas de combustible en cápsulas diseñadas especialmente, en las cuales las varillas se pusieron en estado de avería por un descenso repentino de la capacidad de extracción del calor, obtenido por expulsión de NaK de un anillo alrededor de la varilla. Además se han realizado experimentos para estudiar la ebullición del sodio. Se han desarrollado instrumentos, por ejemplo detectores de vacío, para los experimentos de expulsión de sodio.

Los experimentos se refieren a la clase de accidentes menores que, si no se detectan, pueden hipotéticamente ocasionar accidentes graves como consecuencia de los cuales se vaporizará una parte del núcleo. Este material quedará presente en el aire durante largo tiempo. Para evaluar esta clase de accidentes hay que conocer el comportamiento de los aerosoles en sistemas cerrados. Durante los últimos años se han realizado investigaciones sobre este comportamiento en Petten. Se han dispersado diferentes materiales por medio de la técnica del hilo que explota. Se ha estudiado la formación de aerosoles así como el debilitamiento de la concentración. Los parámetros eran, entre otros, dimensiones de la vasija, temperatura y energía de formación.

1. INTRODUCTION

The work at the Reactor Centrum Nederland (RCN) on sodium-cooled fast reactors is part of the Fast Breeder Project, jointly undertaken by the Federal Republic of Germany, Luxembourg and The Netherlands [1].

One of the RCN tasks concerns an experimental investigation related to accident situations.

The experiments have been divided into three groups: fuel-pin irradiations; sodium boiling experiments; aerosol research.

2. FUEL-PIN IRRADIATIONS

Fuel-pin irradiations were carried out to investigate the mechanisms and consequences of fuel pin failures in the event of disturbances in the sodium cooling and to evaluate the behaviour of fuel and canning under conditions of high heat ratings and high burn-up.

These experiments consisted of: performance tests; short-term irradiations of fuel pins with internal gas pressure and at high sodium temperature (in relation to these experiments, pressure testing of canning tubes has been performed); fuel-pin irradiations in "loss-of-cooling" (LOC) capsules in which the pins can be brought to failure.

All fuel-pin irradiation experiments were carried out in a sodium environment. The fuel-pin dimensions were in accordance with the specifications for the prototype reactor SNR-300.

All experiments were performed in the so called "poolside facility" (PSF) of the High Flux Reactor (HFR) at Petten [2], which at present is operated at 45 MW.

2.1. Performance tests

In this series of experiments, three capsules were irradiated at a heat rating of approximately 650 W/cm and one at 850 W/cm and at burn-ups up to 58 MWd/kg UO_2 .

In Figure 1, the design of a "poolside facility" capsule (PSF) is presented. The fuel pin is placed in the centre of a stainless-steel containment filled with sodium. A shroud tube between the containment wall and the fuel pin promotes natural convection of the sodium along the fuel pin. The shroud tube is also used for attaching the thermocouples. This first containment is assembled in the centre of a second containment, which is filled with a gas. By changing the gas mixture, it is possible to influence the temperature of the sodium. The capsule is water cooled by forced convection.

A summary of the results is given in Table I, from which it can be seen that fuel pin A04 failed at a burn-up of 58 MWd/kg UO_2 . Since the released fission gas inventory could not be measured, it was decided to deduce this value by extrapolation of the results of the other experiments. From the amount of fission gas released, the internal pressure of the pin could be calculated and thus the resulting canning stress (S_p). If one assumes that no relaxation of the thermal stress in the canning (S_{th}) occurred, the total canning stress was $S_p + S_{th}$. As can be seen from Table I, the total canning stress in all experiments remained below the yield strength at operating temperatures. However, considerable interaction between fuel and canning occurred, especially in the pins, with the highest burn-up, so it is likely that this affected the mechanical strength of the canning. Figure 2 shows some details of these interactions.

Electron microprobe analysis of the interaction zones is still in progress. So far, it can be said that the white parts in the interaction zone consist of iron, chromium and nickel, which indicates that these are parts of the stainless-steel canning.

Except around the failure spot of pin A04, in no case were significant changes measured in outer diameter, length and ovality.

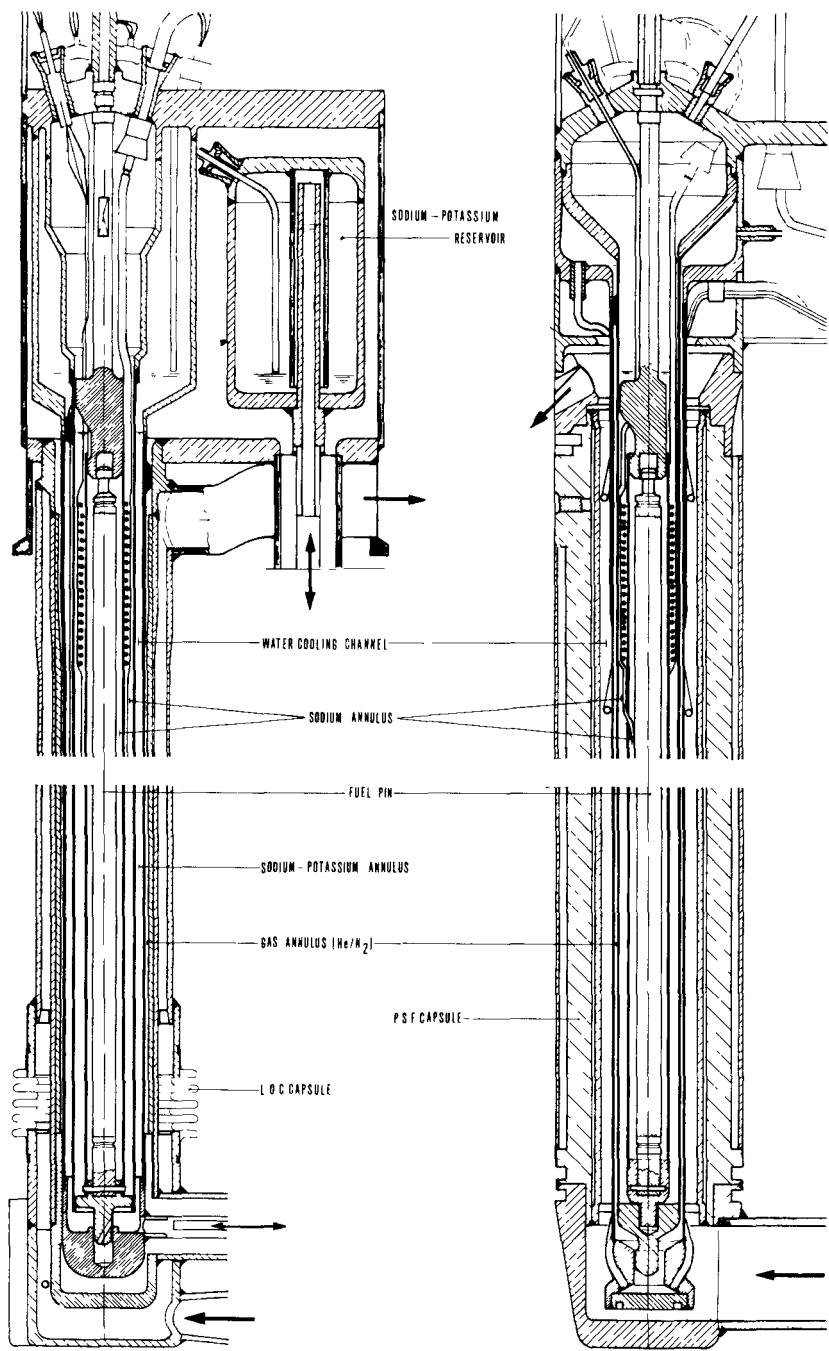


FIG. 1. Irradiation capsules.

TABLE I. IRRADIATION RESULTS OF PERFORMANCE TESTS

Description		A01	A02	A03	A04
Canning material	Werkstoff No.	4961	4961	4988	4988
diameter	mm	6.1/6.7	6.7/6.1	6.0/5.24	6.0/5.24
UO ₂ diameter	mm	6.0	6.0	5.1	5.1
Weight 50% enr. UO ₂	g	70	70	40	40
Gas/fuel volume	%	20	20	50	50
Burn-up	MWd/kg UO ₂	7	20	45	58
Linear power	W/cm	685	850	645	640
Calculated max. av. canning temp	°C	630	600	720	725
Max. centre fuel temp.	°C	2470	2465	2350	2225
Fission gas release	%	28	42	65	(80)
Internal gas pressure	bar	9	31	32.5	(46)
Canning stress due to internal pressure, S _p	kg/mm ²	0.9	3.2	2.2	3.2
Total stress in can- ning, S _p + S _{th}	kg/mm ²	9.8	14.6	14.7	15.7
Unirr. yield strength at indicated canning temperature	kg/mm ²	17.5	19.0	18.6	18.6
Condition after irradiation		Not failed	Not failed	Not failed	Failed

From the first three fuel pins, it was found that the fraction of fission gas released increased from 28 to 68% with an increase in burn-up from 7 to 45 MWd/kg UO₂.

2.2. Short-term irradiations

In preparing the irradiation of fuel pins with internal gas pressure and a high sodium temperature, out-of-pile burst tests on canning tube sections were carried out.

The aim of these burst experiments was to study the mode of failure under simulated abnormal clad operation conditions by pressure testing tube sections, 5.24 mm i. d. × 6.00 mm o. d. × 80 mm heated length. The tube sections were fabricated from 12R72HV tubing (15% Ni, 15% Cr, 1.2% Mo, 1.7% Mn, 0.5% Ti, 0.1% C, balance Fe), and delivered as 15% cold-worked material. The test temperatures were 900, 975 and 1050°C, while the internal gas pressure was varied from 50 to 90 bar. The total gas volume available for energy release during bursting was 15 cm³, being the free gas volume in a full-sized SNR fuel pin. A description of the test apparatus and the experiments is given in Ref. [3].

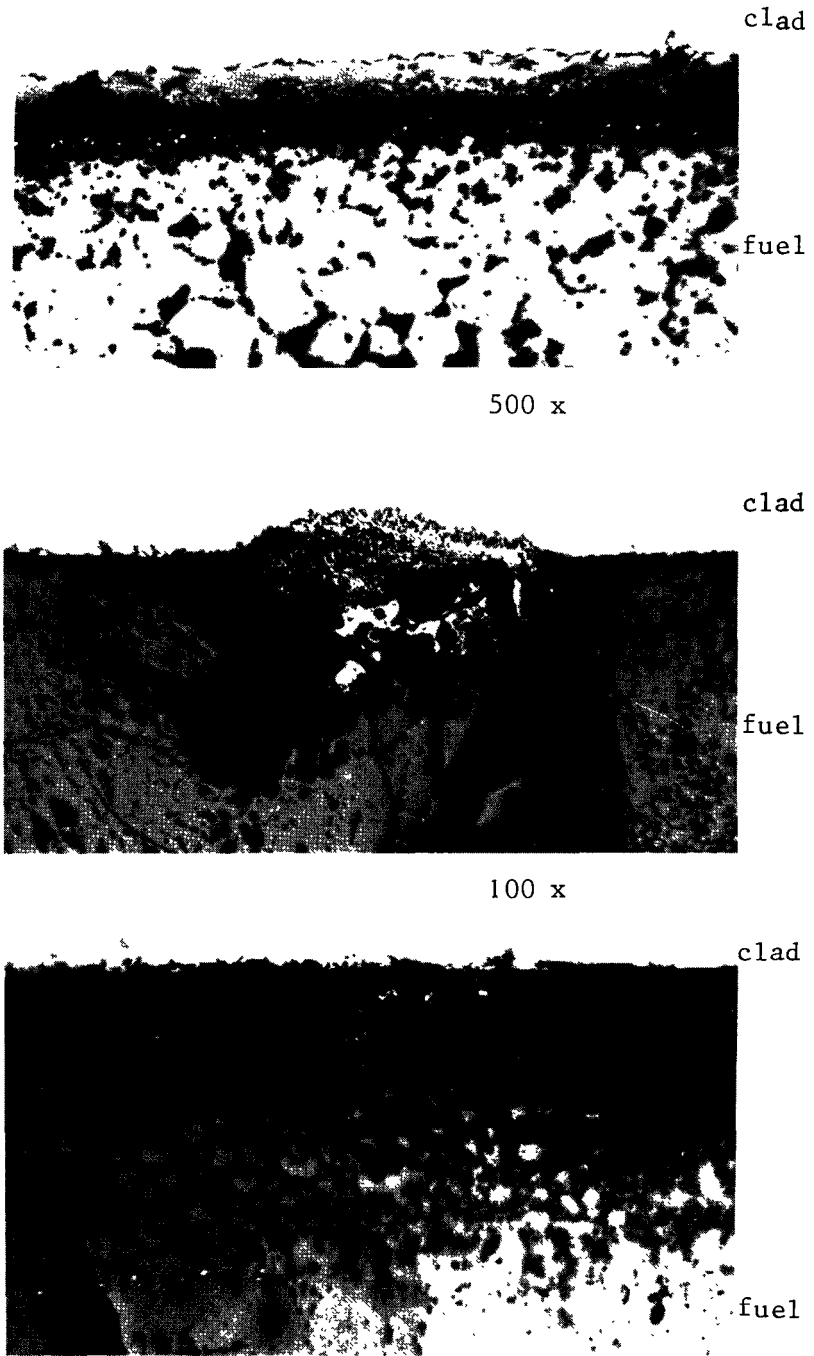


FIG. 2. Examples of the interaction zone between fuel and cladding of pins with a high burn-up ($\times 500$).

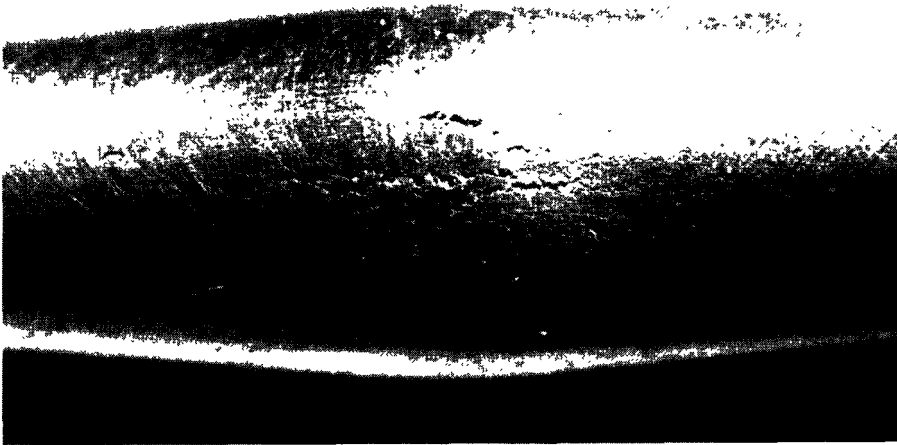


FIG. 3. Outer surface appearance, after polishing, of a 12R72HV tube, pressure tested up to failure after 37 min at 90 kg/cm² and 975°C.

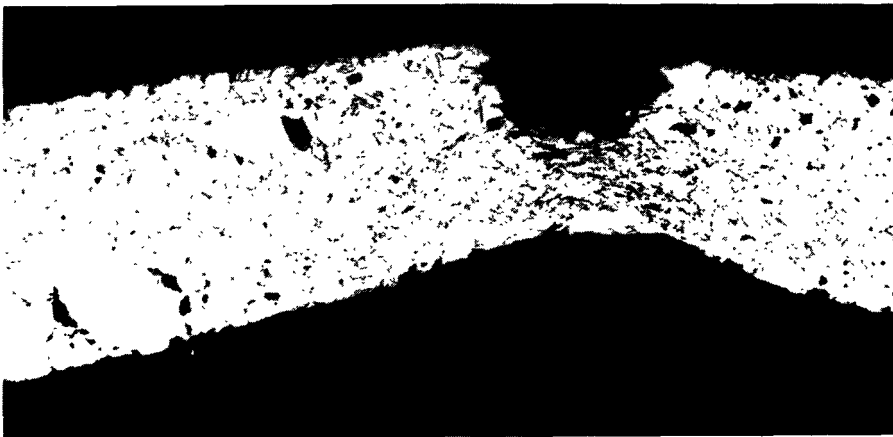


FIG. 4. Metallographic structure of the tube wall presented in Fig. 3, in the vicinity of the failure (× 100).

The structural and surface appearance of a tube section after bursting is presented in Figs 3 to 5. Bursting of the tubes apparently occurred by gradual, intergranular cracking, with the main cracks originating from the outer surface. Because of the small gas volume behind the burst, equivalent to that in a normal fuel pin, neither violent bursting nor extraordinary expansion of the cladding occurred at the moment of failure.

At 1050°C, non-violent bursting of the canning also occurred. It can thus be expected that failure of the fuel pins during the short-term irradiation will not lead to gross expulsion of fuel particles.

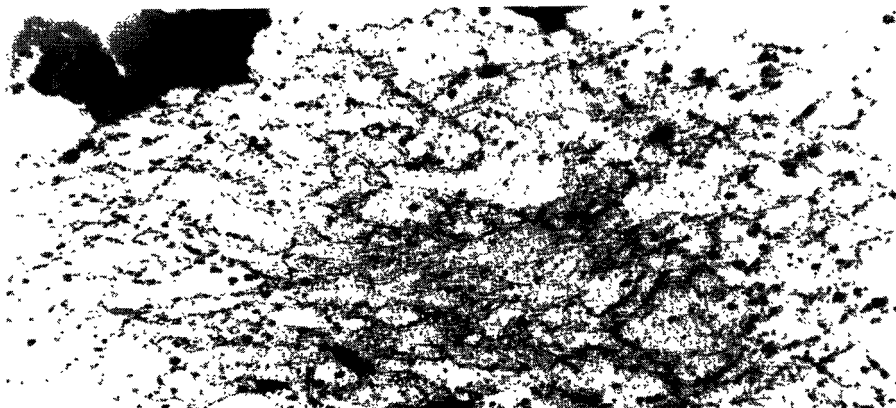


FIG. 5. Metallographic structure of the tube wall presented in Fig. 4, in the bottom of the crack ($\times 500$).

Based on the results of these tests, the combination of internal pressure and average canning temperature for the irradiation experiments will be chosen so that failure may occur within one hour.

2.3. Loss-of-cooling experiments

The main purpose of these experiments is to investigate the consequences of a fuel-pin failure, particularly with respect to liquid fuel/sodium interaction. In Ref. [1], a number of possible consequences are presented.

For these experiments, a "loss-of-cooling" (LOC) capsule was developed which is shown in Fig. 1. As can be seen, this capsule is a modified PSF capsule. The fuel pin is surrounded by sodium which is enclosed in a containment, a shroud tube being placed between the fuel pin and the containment. The sodium containment is surrounded by a sodium-potassium mixture enclosed in a second containment, which is connected to an outside reservoir. The NaK containment is, in turn, surrounded by a gas-filled annulus, which serves as a variable heat barrier and provides a double containment around the NaK. The capsule is cooled by forced convection with water.

The "loss-of-cooling" condition is simulated by expelling the NaK from the annulus around the sodium containment to the reservoir. In this way the radial heat transport will decrease rapidly, resulting in a very fast increase of the sodium, canning and fuel temperatures. The appearance of eventual pressure pulses can be measured and registered by pressure transducers which are connected to the sodium containment. During the "loss-of-cooling" period the fuel pin remains surrounded by sodium.

It should be noted that the PSF capsule and LOC capsule can be moved to and from the reactor core during reactor operation.

The experiments were started with an irradiation of 30 minutes under nominal conditions (maximum 600 W/cm and an average canning temperature of 540°C).

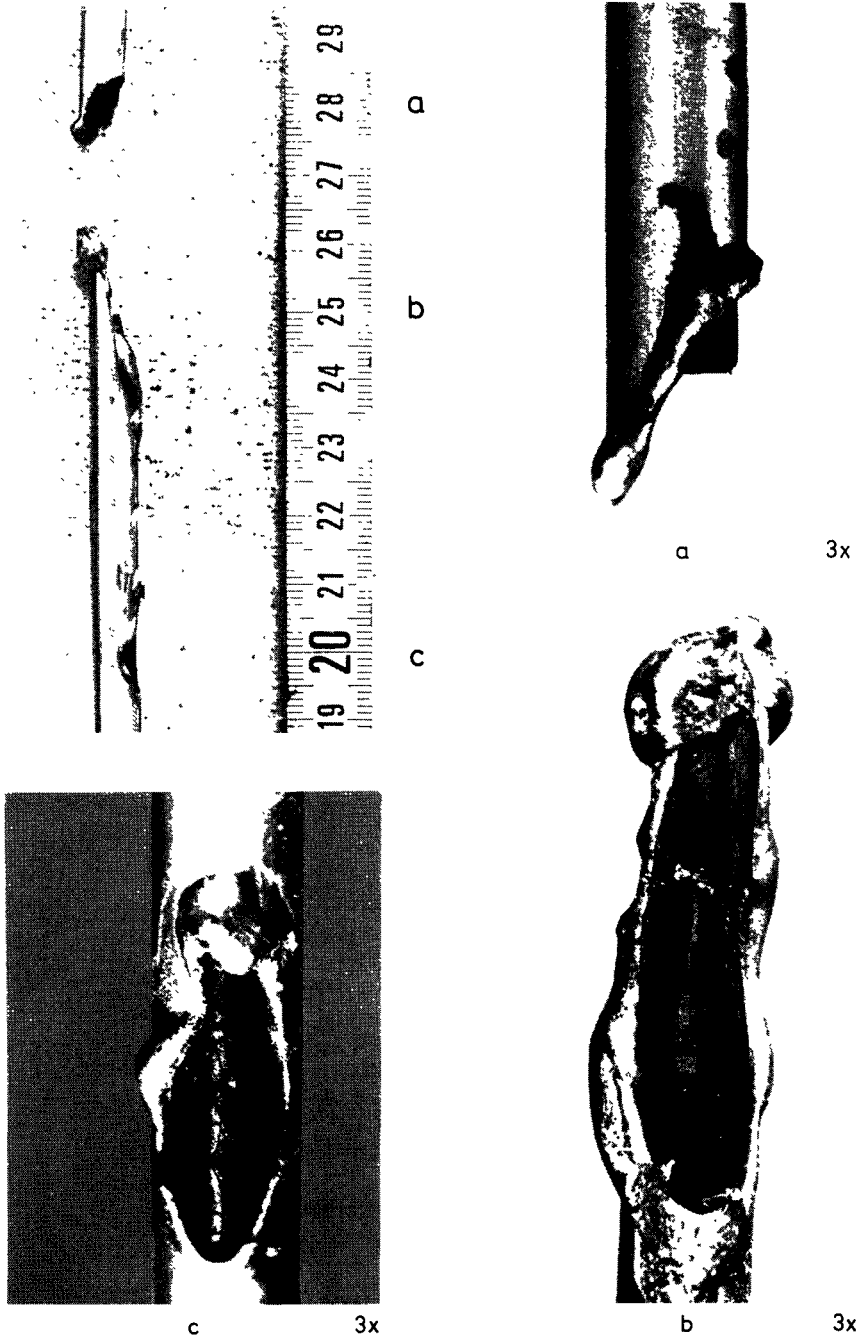


FIG. 6. View of a failed "loss of cooling" pin and some details of the failure spots.

The first experiment showed that, within a timespan of 5.9 seconds between the start of the "loss of cooling" and the reactor scram, the fuel pin failed. An X-ray of the capsule indicated that the surrounding containments were not damaged. The temperature increase of the sodium during the "loss-of-cooling" period was roughly 100°C/s and the highest measured sodium temperature was 1040°C .

Under these circumstances, the canning temperature should not reach the melting point if sodium boiling does not occur. However, post-irradiation examination revealed that fuel-pin failure occurred by melting of the canning at two places. This means that, at those places, the sodium should have boiled. Figure 6 gives a view of the pin at the failure sites, and it can be seen that no fuel has disappeared. The fuel column between details b and c was intact; however, during the dismantling operation, it broke into three pieces.

Future experiments will be with extended delay times between the start of the "loss of cooling" and the reactor scram.

3. SODIUM BOILING EXPERIMENTS

In the context of this paper, the reasons for research on sodium boiling are obvious. The RCN sodium boiling program [4] covers: the mode of sodium boiling in a sub-channel as influenced by the hydraulic diameter, sodium velocity, heat flux and pressure; the growth of a single vapour bubble in a 12-rod bundle as a function of heat flux, pressure and sodium superheat (the sodium superheat can be varied by means of an electrical ignition).

The above parameters will be varied around SNR conditions, while the sodium purity will be monitored.

Some investigations on the mode of boiling were carried out in a natural convection loop [5]. The hydraulic diameter of the annular channel was 4.2 cm.

The heater elements developed by RCN are essential to this program. They consist, in principle, of three concentric layers: the electrode, an isolation layer and the canning material. The electrodes and cannings are both of molybdenum. Electrical isolation is achieved with boron-nitride. Some further details of the used heaters are: outer diameter $\phi = 0.75$ cm, heated length $l = 10$ cm.

The heaters were instrumented with two thermocouples in the hollow electrode, one embedded in the canning and five at radial intervals of 0.2 cm in the sodium. The heat fluxes were varied between $300\text{--}650$ W/cm^2 at a pressure of 1.0–1.6 bar.

The sodium velocity was very low (1–10 cm/s), so the conditions might simulate a pump failure in a fast sodium-cooled reactor. The annular velocity profile was disturbed by free convective eddies near the heater wall. This resulted in high temperature fluctuations ($60\text{--}70^{\circ}\text{C}$). The eddies reached 0.4 cm into the sodium.

As a result of the onset of boiling, the wall superheat fluctuated from 20°C to 45°C . The influence of the heat flux ($300\text{--}450$ W/cm^2) and pressure (1.0–1.6 bar) could not be distinguished within this range of superheats.

In these experiments with inlet sub-coolings from $90\text{--}130^{\circ}\text{C}$, sub-cooled nucleate boiling with some "bumping" existed. The wall temperature fluctu-

ations fell below 5°C. The bubble departure caused temperature fluctuations of 80°C in the sodium.

At the end of this experimental series, the burn-out heat flux was determined at 660-685 W/cm² with a system pressure of 1.4 bar, a bulk temperature of 775°C and a sodium velocity of 7 cm/s.

Calculations with a computer code CACES [6] showed temperature differences up to 36°C in a sub-channel of a rod bundle at SNR conditions with reduced flux.

The experiments to classify the mode of boiling at SNR and other conditions will be continued.

4. AEROSOL RESEARCH

The previously mentioned experiments concerned the class of minor accidents which, if undetected, may hypothetically lead to a major accident, the Design Basis Accident.

As a result of such an accident, a certain fraction of the core will be released as air-borne material into the first containment [7]. The Aerosol Model, which introduces the concept of natural instability of aerosols in the safety evaluation, was applied to the SEFOR reactor [8] but rejected. Häfele et al. [9], however, have shown that the Aerosol Model is a solution of the safety problems and will hence save construction costs.

The aerosol experiments have to simulate aerosol formation (by explosive vaporization and condensation) and the influences on the fate of the aerosol (atmosphere, dimensions and temperature of the containment). Moreover, rules must be obtained to permit extrapolation of the small-scale laboratory experiments to reactor scale.

4.1. Material and methods

The aerosols are generated in closed tanks by means of the exploding wire technique [10], enabling simple variation of the specific energy of aerosol formation. Table II summarizes the characteristics of the aerosol tanks used. For about 100 hours, the decay of the concentration of aerosol mass and aerosol particle number is measured. Mass concentrations are determined by using neutron-activated aerosol material and counting filter loads. Particle number concentration was measured either by means of a condensation nuclei counter or by counting the aerosol particles on an electron micrograph of a known sample taken by an electrostatic precipitator [11].

TABLE II. CHARACTERISTICS OF AEROSOL TANKS

	Volume	Height	<u>Internal surface area</u> volume
100	1.2 m ³	1.5 m	5.2 m ⁻¹
200	1.0 m ³	2.0, 1.0 or 0.5 m	7.0 m ⁻¹
ENAK	20 m ³	4.2 m	2.1 m ⁻¹

4.2. Results and discussions

Figure 7 shows the influence of the specific formation energy (E_{spec}) on the decay of the mass concentration of a gold aerosol in the 100-tank. The significant decrease in aerosol stability with decreasing E_{spec} can be related to the size distribution of the so-called primary particles formed by the explosion (Fig. 8). The size distribution of these particles is log normal with a standard deviation of 1.5, irrespective of E_{spec} . The geometric median diameter, however, decreases when E_{spec} is increased. Although the size distribution is soon disturbed by coagulation, E_{spec} retains its influence on long-term aerosol behaviour. Since the E_{spec} of the accident aerosol is assumed to be about 5 kJ/g, the experiments are a good simulation of the accident.

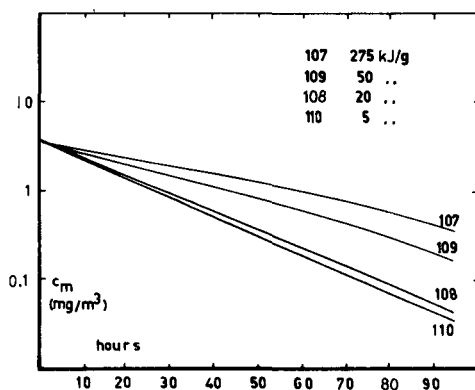


FIG. 7. Mass concentration as a function of time for gold aerosols generated with different specific energies.

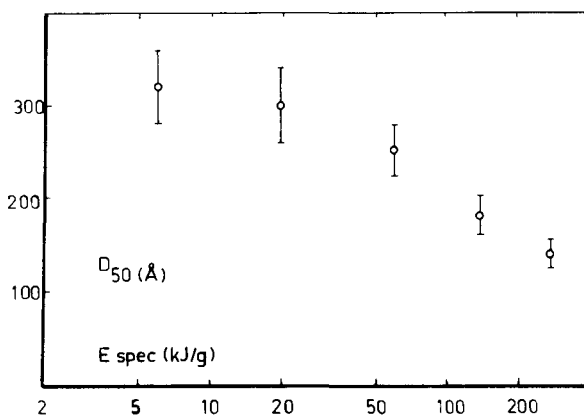


FIG. 8. Geometric median diameter (D_{50}) of primary gold particles formed by explosion of gold wires with different specific energies.

The influences of the tank dimensions on aerosol behaviour were studied. The results lead to a theory of a steady state for the aerodynamic size distribution of enclosed aerosols due to the combined action of diffusion and gravity deposition to the tank walls [12, 13]. Figures 9 and 10 give the decay of the mass concentration and number concentration of a gold aerosol for a wide variety of tank dimensions. It can be concluded that the aerosol disappearance is not a unique function of the height and volume alone, but also of the internal surface area of the tank.

Experiments have been conducted to study the behaviour of mixed aerosols. In a mixed aerosol of gold and copper, produced by simultaneous explosion of two wires, a constant gold-to-copper ratio was found during a period of more than 30 hours. The same was observed for a mixed aerosol of sodium with minor amounts of gold or strontium oxide. The experiments

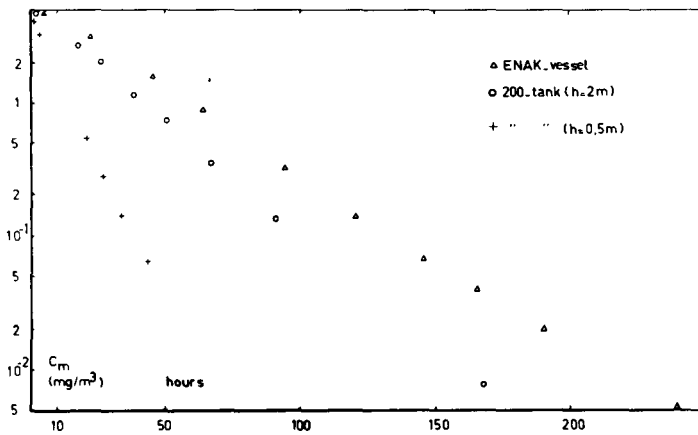


FIG. 9. Mass concentration decay of gold aerosol in different aerosol tanks. $E_{\text{spec}} = 20 \text{ kJ/g}$.

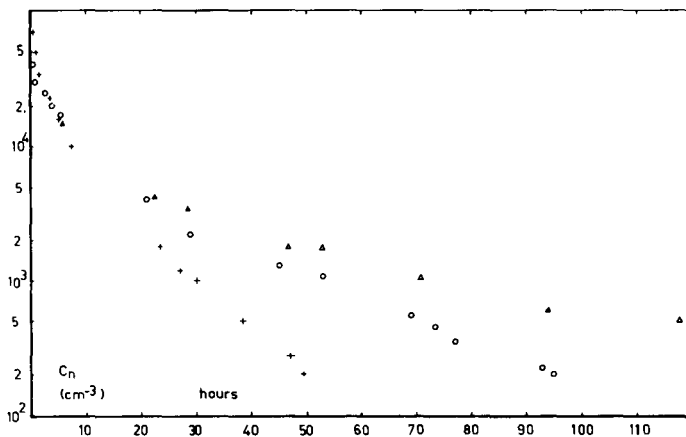


FIG. 10. Decay of particle number concentration of gold aerosol in different aerosol tanks. $E_{\text{spec}} = 20 \text{ kJ/g}$. For explanation see insert of Fig. 9.

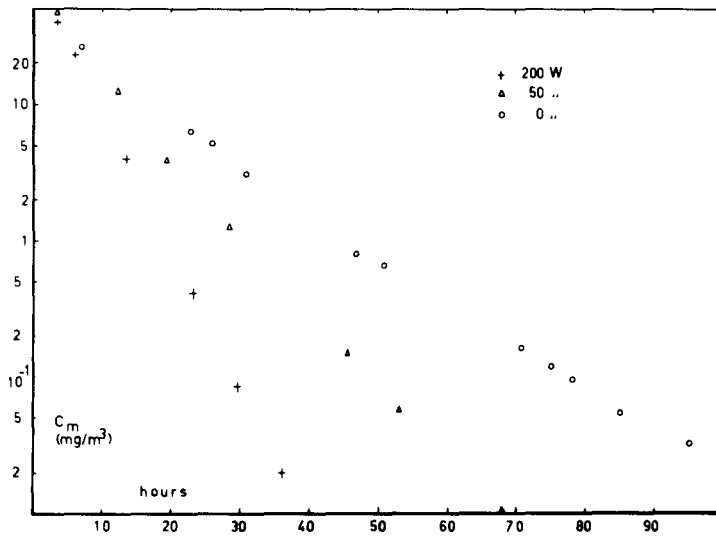


FIG. 11. Mass concentration decay of gold aerosols in 100 tank. At the centre of the floor a heater is placed. The formation is identical: 80 mg/m^3 and $E_{\text{spec}} = 20 \text{ kJ/g}$.

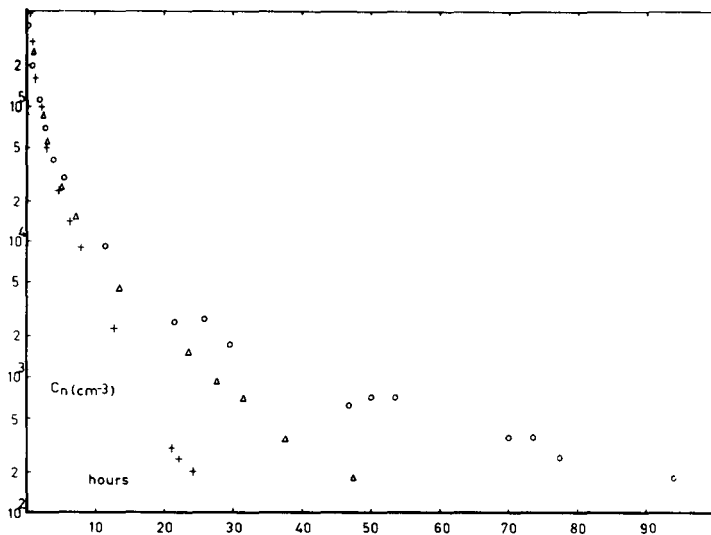


FIG. 12. Decay of particle number concentration of gold aerosols in 100 tank. For further explanation see caption and insert of Fig. 11.

with sodium aerosols are carried out in nitrogen, but there is very little effect of oxidation on aerosol behaviour. From these experiments, it can be concluded that the behaviour of dangerous accident aerosols of fission products or plutonium is determined by the major air-borne component in the containment, namely sodium in the case of a liquid-metal fast breeder reactor (LMFBR).

The temperature distribution in the containment after the accident is determined by the decay heat production of the core material deposited on the floor. The influence of such heterogeneous space heating on the enclosed aerosol was studied. Figures 11 and 12 depict the great influence of a heat source at the centre of the floor. Observation of the aerosol deposition in a lucite box with analogous heating revealed enhanced deposition near the heater, leading to the conclusion that thermophoresis is responsible for the strong deposition in case of heterogeneous heating. Heating of the whole floor surface showed the same effects, but the floor remained free of aerosol. This means that sedimentation is strongly hindered by the temperature gradient near the floor. Until now, it has been accepted that accident aerosols are mainly removed by sedimentation [14,15], but new extrapolation rules must be developed if thermophoresis is the natural removal mechanism.

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REGULATORY PERSPECTIVES AND EMPHASIS, AND SAFETY EXPERIENCE FOR NUCLEAR POWER REACTORS

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Abstract—Résumé—Аннотация—Resumen

REGULATORY PERSPECTIVES AND EMPHASIS, AND SAFETY EXPERIENCE FOR NUCLEAR POWER REACTORS.

The US statutes provide that nuclear facilities may be constructed and operated only under license authorization by the USAEC. Through the license authorization process, USAEC exercises regulatory surveillance over the design, construction, and operation of each nuclear facility throughout its lifetime. The purposes of this control are to ensure that nuclear facilities are so designed, constructed, and operated that routine radioactive effluents will be maintained within permissible limits, that there is adequate protection against accidental releases of radioactivity, that due attention is given to protection against other possible adverse environmental effects, and that operation is not inimical to the common defense and security.

Since the first nuclear-power reactor began operation in 1957, an extensive system of procedures has to ensure that design requirements to accomplish the objectives listed above have been factored into the facility design and to provide a system of review and inspection to verify that these requirements are indeed built and appropriately used. This system is still evolving, and in recent years the underlying principles and concepts of safety protection have coalesced into the general practice of 'defense in depth'. The dual objectives of this system are (1) prevention of accidents and (2) controlling the consequences should an accident occur. Recent trends include (1) a more uniform application of accident analysis; (2) new safety protection features resulting from follow-through on accident analysis; (3) development and application of a quality-assurance system, (4) increased attention to minimization of adverse environmental effects; and (5) revised procedures to accommodate the growing number of power-plants and the increased interest in public participation.

Some 20 nuclear-power plants, with a combined capacity of 7551 MW(e), are now licensed for operation. An additional 55 plants have been granted construction permits, and applications for some 31 plants are under review. This paper discusses some of the experiences gained in the construction and operation of these plants. Summaries are provided on problems encountered in operation, on the record of routine effluents released to the environment, and on radiation-exposure data for operating personnel.

PERSPECTIVES ET CARACTERISTIQUES DE LA REGULATION DES CENTRALES NUCLEAIRES ET EXPERIENCE EN MATIERE DE SECURITE.

Les règlements fédéraux des Etats-Unis d'Amérique stipulent que les installations nucléaires ne peuvent être construites et exploitées qu'avec une autorisation accordée par l'USAEC. Par cet intermédiaire l'USAEC exerce une surveillance régulatrice sur le projet, la construction et le fonctionnement de chaque installation nucléaire tout au long de son existence. Les objectifs de ce contrôle sont de s'assurer que les installations nucléaires sont projetées, construites et exploitées de telle sorte que les effluents radioactifs évacués ordinairement ne dépassent pas les limites permises, que le public est correctement protégé contre les risques de libération accidentelle de radioactivité, que toute l'attention nécessaire est portée à la protection contre d'autres effets possibles néfastes pour l'environnement et que l'exploitation ne porte pas atteinte à la défense commune et à la sécurité des Etats-Unis.

Depuis que la première centrale nucléaire a commencé à fonctionner, en 1957, un système étendu de procédures a été mis au point pour contrôler que les conditions imposées par les objectifs indiqués ci-dessus ont été respectées dans le projet d'installation. Il tend également à mettre en place un système d'inspection visant à vérifier que les dispositifs figurant dans le projet ont été effectivement construits et sont correctement utilisés. Ce système est encore en cours de développement et, ces dernières années, les principes et les conceptions de la protection et de la sécurité ont été rassemblés dans un règlement

de «défense en profondeur». Les deux objectifs de ce système sont: 1) de prévenir les accidents et 2) de limiter les conséquences de tels accidents dans le cas où ils se produiraient. Ce système fait actuellement l'objet de nouveaux développements dans les directions suivantes: 1) application plus systématique des analyses d'accidents; 2) définition de nouvelles conditions de sécurité à la suite de la poursuite de ces analyses; 3) développement et application d'un système assurant la bonne qualité des réalisations; 4) renforcement de l'attention portée à la minimisation des effets néfastes pour l'environnement; et 5) révision des procédés rendue nécessaire par l'augmentation du nombre de centrales nucléaires et l'intérêt accru du public.

Quelque 20 centrales nucléaires d'une puissance totale de 7551 MW(e) ont actuellement reçu une licence. Cinquante-cinq autres installations ont reçu le permis de construire et des demandes de licence pour 31 centrales sont à l'étude. Le mémoire discute l'expérience acquise dans la construction et le fonctionnement de ces installations. Les problèmes ayant trait au fonctionnement, à l'enregistrement des effluents normalement libérés dans l'environnement et aux données relatives à l'exposition aux radiations du personnel des centrales sont résumés.

ЗАКОНОДАТЕЛЬНЫЕ ПЕРСПЕКТИВЫ И ОСНОВНЫЕ ПРОБЛЕМЫ, А ТАКЖЕ ОПЫТ БЕЗОПАСНОЙ ЭКСПЛУАТАЦИИ ЯДЕРНЫХ ЭНЕРГЕТИЧЕСКИХ РЕАКТОРОВ.

Законы США предусматривают, что сооружение и эксплуатация ядерных установок может осуществляться только при наличии лицензии, выданной КАЭ США. Путем выдачи лицензий КАЭ США осуществляет законодательный надзор над конструированием, сооружением и эксплуатацией каждой ядерной установки в течение всего срока ее службы. Целью этого контроля является обеспечение того, чтобы ядерные установки конструировались, сооружались и эксплуатировались так, чтобы обычные радиоактивные отходы удерживались в допустимых пределах и чтобы была обеспечена соответствующая защита от случайных выбросов радиоактивных веществ, а также чтобы уделялось должное внимание защите от других возможных неблагоприятных воздействий на окружающую среду и чтобы эксплуатация установок не причиняла вреда общей защите и безопасности.

С момента пуска в эксплуатацию в 1957 году первого ядерного энергетического реактора была разработана обширная система процедур, с тем чтобы конструктивные требования, направленные на реализацию вышеуказанных задач, были учтены при проектировании установок, а также чтобы была создана система надзора и инспекций в целях проверки, что эти требования действительно учтены и соответствующим образом используются. Эта система продолжает развиваться, и за последние годы основные принципы и концепции обеспечения безопасности были объединены в общую практику "защита в глубину". Эта система имеет две цели: 1) предотвращение аварий, которые могут привести к выбросу продуктов деления и 2) ограничение последствий, если произойдет такая авария. Наиболее важными современными направлениями в этой области являются: 1) более единообразный и тщательный анализ аварий; 2) новые способы обеспечения безопасности, ставшие возможными в результате проведения тщательного анализа аварии; 3) развитие и применение системы обеспечения качества; 4) повышенное внимание к вопросам уменьшения неблагоприятных воздействий на окружающую среду; и 5) пересмотр процедур в соответствии с увеличением числа энергетических установок и возросший интерес общественности к решению данных вопросов.

В настоящее время выданы лицензии на эксплуатацию 20 атомных электростанций общей мощностью 7551 МВт (эл). Выданы разрешения на строительство еще 55 станций. В стадии рассмотрения находятся заявки на строительство 31 станции. В данном докладе рассматривается приобретенный опыт в деле строительства и эксплуатации таких станций. Перечисляются также проблемы, с которыми приходится сталкиваться в ходе эксплуатации, приводятся зарегистрированные данные об обычных сбросах отходов в окружающую среду, а также данные о дозах облучения работающего персонала.

PERSPECTIVAS Y CARACTERISTICAS REGLAMENTARIAS DE LAS CENTRALES NUCLEARES Y EXPERIENCIA EN MATERIA DE SEGURIDAD.

Según la legislación federal de los Estados Unidos de América, las instalaciones nucleares sólo se pueden construir y explotar previa autorización de la USAEC. Por medio de los trámites de autorización, la USAEC ejerce la fiscalización reglamentaria del proyecto, construcción y explotación de cada instalación nuclear durante toda su vida útil. Los objetivos de esta fiscalización son velar por que las instalaciones nucleares se proyecten, construyan y funcionen de modo que los efluentes radiactivos habituales se mantengan dentro de límites admisibles, por que el seguridad público esté adecuadamente protegido contra escapes fortuitos de radiactividad, por que se atienda debidamente a la protección contra otros posibles efectos adversos sobre el medio ambiente, y por que la explotación no redunde en menoscabo de la defensa y seguridad nacionales.

Desde 1957, año en que entró en funcionamiento la primera central nuclear, se ha elaborado un extenso sistema de normas a fin de garantizar que las instalaciones cumplan las condiciones necesarias para conseguir los objetivos enumerados anteriormente y, también, a fin de crear un sistema de inspección para comprobar que las instalaciones responden efectivamente a tales condiciones y las cumplen debidamente. El sistema continúa desarrollándose, y en los últimos años los principios y conceptos fundamentales de seguridad han cristalizado en la práctica general de «defensa escalonada». Los dos objetivos de este sistema son: 1) prevenir los accidentes y 2) limitar las consecuencias en el caso de que se produjeran dichos accidentes. Entre las tendencias recientes cabe citar: 1) más uniformidad en la aplicación de los resultados de los análisis de accidentes; 2) nuevas medidas de seguridad a resultas del análisis continuado de esos accidentes; 3) perfeccionamiento y aplicación de un sistema que garantice la calidad, 4) creciente atención para minimizar los efectos adversos sobre el medio ambiente; 5) modificación de las formalidades para responder al aumento de centrales eléctricas, y al creciente interés del público.

Actualmente hay autorizadas para su funcionamiento 20 centrales nucleares con una potencia total de 7551 MW(e). Se han concedido autorizaciones para la construcción de otras 55 centrales y se están examinando solicitudes para otras 31 centrales. La memoria trata de la experiencia adquirida en materia de reglamentación mediante la construcción y funcionamiento de estas centrales. Se presentan recapitulaciones relativas a los problemas que plantea su explotación, al registro de las descargas habituales de efluentes en el medio ambiente, y a los datos sobre exposición a las radiaciones del personal dedicado a la explotación de las centrales.

1. INTRODUCTION

The formal procedures for safety reviews by the United States Atomic Energy Commission (USAEC) of applications for construction permits and operating licenses for nuclear power reactors have been described a number of times [1, 2]. These procedures, initiated in the late 1950's, have been evolving with time as more experience and changing conditions have come about. Similarly, with many larger reactors of more advanced design, new developments have occurred in the scope and depth of the safety reviews.

The purpose of this paper is to describe the developments of the last few years with respect to procedural changes and safety questions of special interest, and to provide a brief summary of the experience with operation of licensed nuclear power plants in the United States of America.

2. RECENT TRENDS AND EMPHASIS IN SAFETY REVIEWS

Early safety reviews in the United States were largely concerned with the neutronics of reactor cores, power stability and reactivity excursion control, whereas in current reviews in addition to consideration of these matters the greatest emphasis is placed on assurance of the means for core cooling under all circumstances. Earlier safety reviews of light-water-cooled reactors also tended to emphasize the concept of a maximum credible accident and to evaluate the adequacy of the reactor site or the integrity of its containment in judging the acceptability of its design and location. At the present time the AEC considers in its reviews a number of design basis accidents and recognizes that isolated sites by themselves or low leak-rate containment provisions by themselves may not provide adequate protection.

Design basis accidents are used in assessing the overall safety of a plant in several ways [3]. First, in concert with the requirements of 10 CFR Part 100, Reactor Site Criteria (of the AEC's regulations), offsite doses are calculated that might result from various accidents of very low probability.

Such doses calculated using a conservative model are required to be within the limits of the Part 100 guidelines, and, for relatively less improbable incidents, the calculated doses are required to be well within these guidelines. Assurance that calculated potential offsite doses are within the guidelines defines, in effect, the performance criteria that the plant design will need to meet. In analysing particular design basis accidents an examination is made of fission product retention by inherent processes and engineered safety features. Consideration of these results determines the reliability required in the initial design, in the protective features and in the systems that mitigate the consequences of accidents. Also, it forces an examination of the conservatism of the assumptions underlying the design and performance specifications.

The developments described above have led to two rather basic concepts in the overall consideration of the requirements for an adequately safe nuclear industry. First is the concept of quality assurance, and all that the term now implies. Second is the concept of defense-in-depth [4], which finds expression in various ways.

Quality assurance in reactor design, construction and operation has been defined and discussed in several AEC documents [5,6]. In brief, however, what is necessary is a well-defined procedural system, established and carried out by the reactor owner-operator, that produces a very high degree of assurance that the concept, design, procurement, fabrication, test and operation of the plant all meet exacting standards of quality — standards of quality not generally met in the non-nuclear industry. The most important ingredient in a satisfactory quality assurance program is the provision for audit of all functions of significance to safety, integrity and plant performance at each stage of development by persons or groups independent of those with line responsibility for getting the job done. This is not to be taken to imply that quality is achieved by inspection; it is not. Quality is achieved by the accomplishments of those whose responsibility it is to get the job done. The concept of quality assurance is based on the premise that those having responsibility for performing an activity will perform it more responsibly if there is provision for an independent verification that the activity was performed at the required level of quality. Such a quality assurance system may be considered as an application of defense-in-depth of the performance of people, with two successive levels: the line organization and the independent licensee audit.

To give some guidance to the actions necessary to develop an adequate quality assurance program, the AEC published in August 1970, as Appendix B to 10 CFR Part 50, 'Quality Assurance Criteria for Nuclear Power Plants'.

Another way of thinking about defense-in-depth is to consider the degree of independence and reliability of the three different functional parts of the overall plant design. First, one may consider the reactor itself, its inherent safety features and the normally functioning systems. Second, one may consider the 'external' control and safety features provided to minimize the chance for deviation from design performance. Third, one may consider the engineered safety features and isolation of the site that are provided to mitigate the consequences of any accident that might occur.

Defense-in-depth in design also may be thought of in the following way. One may examine and evaluate independently the several barriers to the transport of fission products out of the fuel itself and into the coolant, then out of the primary coolant system into the containment system, then out of

the containment and confinement systems into the environment, and finally through whatever critical pathways may exist to the final exposure of man. In addition to evaluating the number and independence of the barriers, one also should examine the reliability and effectiveness of each one. In assessing the ultimate risk to the public, a definition also is needed of the source term, i.e. the probability and magnitude of release of fission products from the fuel for each postulated initiating mechanism.

For release of fission products from the fuel, the AEC continues to use the release fractions described in TID 14844, 'Calculation of Distance Factors for Power and Test Reactor Sites'. When this document was developed in 1962, the release fractions were estimated conservatively on the basis of what might be expected to be released if the reactor core should melt. Melting of a substantial fraction of the core of a current-design light-water power reactor conceivably could lead to rupture of the reactor pressure vessel and then in turn to rupture of the containment system, resulting in an unacceptably large release of activity to the environment. Consequently, an unusually high degree of effort is expended to assure the reliability of the reactor vessel, the primary system and the normal and emergency cooling water systems. Nevertheless, the design bases for engineered safety features to protect against a design basis loss-of-coolant accident¹ including containment structures, and the calculated doses to people off-site, are based on the fission product release fractions originally postulated for a melted core.

As more and larger reactors are being built and operating experience is being accumulated, the AEC has given active and increased attention to understanding the causes and decreasing the likelihood of failure of systems and components whose failure could lead to undesirable consequences [7]. While redundancy of systems and components has been a common goal of the industry from early times, and is well developed in principle, for example, in the IEEE Standard #279 for protection system instrumentation, more emphasis is being given to the assessment of independence and functional diversity of all systems and components important to safety. The objective is to identify, and to protect against, potential common modes of failure that could negate all of the supposedly independent redundant systems. To this end the AEC has requested each of the light-water reactor suppliers in the United States to conduct studies of these potential failure modes and their consequences. Related to this are similar studies of the frequency and consequences of transients anticipated during the lifetime of a plant, where necessary protective action is required, but might fail to be provided. (This might include, for example, a turbine trip with failure of the reactor to scram.)²

Recognizing that more specific guidance for reactor designers and operators was necessary to achieve the goals mentioned above, the AEC has provided various types of guidance in the last several years and is working on extensions of these at the present time. Brief descriptions are given below.

¹ Equivalent to the double-ended rupture of the largest pipe connected to the reactor pressure vessel.

² A review of many of the important technical safety issues involved in safety analysis and evaluation by the AEC of light-water power reactor facilities is provided in Ref. [7].

2.1. General design criteria

The AEC has been developing General Design Criteria for nuclear power plant construction permit applications since 1964. While these have been directed toward, and primarily developed on the basis of experience with, light-water-cooled reactors, it was intended that they be adaptable to all types of power reactors. The criteria were first issued for interim guidance in 1965 and re-issued in 1967. Following extensive discussion with industry representatives, 55 criteria were published as mandatory requirements in Appendix A to 10 CFR Part 50, 'General Design Criteria for Nuclear Power Plants', made effective on March 21, 1971. These criteria will give guidance to the industry on the AEC design requirements; they should result in more complete applications being filed and, therefore, are expected to be of assistance in reducing staff time and effort required in the review and evaluation of proposed nuclear power plant designs.

2.2. Safety guides

In November 1970 the AEC issued the first few in a new series of safety guides. Each safety guide consists of an acceptable resolution to a safety issue where it has not yet been determined that a particular solution should be made a formal AEC requirement.

Compliance with safety guides is not required. Solutions different from those set out in the guides will be acceptable if they are judged to provide an acceptable alternative to the solution described in the guide.

Safety guides, and related work on codes and standards, are discussed in more detail in the paper by Ramey et al. [8].

2.3. Criteria for radioactivity in effluents

The AEC's radiation protection standards, as developed for regulatory purposes, were first made effective as 10 CFR Part 20 in December 1956. These standards have provided limits in terms of maximum permissible doses to people or maximum permissible concentrations of radioactivity in liquid and gaseous effluents. The standards were developed to be consistent with the recommendations of the International Commission on Radiological Protection (ICRP) and the National Committee on Radiation Protection and Measurements (NCRP). These standards also were consistent with the advice of the President's Federal Radiation Council, established later.

Although quantitative limits have been agreed upon by the standards making bodies, noted above, for maximum doses permissible to both radiation workers and to the public, and for concentrations of radioactivity in effluents to restricted and to unrestricted areas, these bodies always have recommended in addition that any radiation may be harmful and that the benefits of radiation should be balanced against the risks. The International Commission on Radiological Protection, as early as 1958, and the National Committee on Radiation Protection and Measurement and the Federal Radiation Council, at other times, recommended that exposures from man-made sources of radiation be kept 'as low as practicable', an expression of the same principle in different words.

Within the last few years, however, several developments have pointed toward the need for more concrete means for keeping radioactive releases

and doses to people 'as low as practicable'. Operating experience with light-water-cooled power reactors, especially those in operation in 1967 and later, has shown that in general radioactive releases have been only small percentages of the limits of the standards. There has been a pronounced increase in the interest of the public and the government to minimize the deleterious effects of all industrial activities upon the environment. Finally, through recent technological developments, practical means now exist for limiting releases of radioactivity from nuclear power plants to the environment to levels that are essentially negligible.

In recognition of these developments, the AEC has carefully examined the changes that could be made in its regulations to assure that the radioactivity in effluents will be as low as practicable. Such changes would need to take into account the nature of fuel element performance and other requirements to provide reliable and dependable sources of electrical power, which make it necessary to preserve a certain amount of flexibility within the standards. In April 1970 the AEC announced its intent to require by formal regulation that radioactive releases from nuclear plants be as low as practicable and indicated that it would study, together with industry, the feasibility of setting quantitative limits to achieve this goal. After extensive discussions with industry representatives, with conservationists and with other interested bodies, the AEC published on December 3, 1970 its new revision to Part 20, to implement the concept of 'as low as practicable', by giving quantitative guidance on permissible levels of radioactivity in effluents for which equipment should be designed and action levels for corrective measures. This change of rules became effective on January 2, 1971.

2.4. Monitoring and reporting radioactive effluent experience

AEC rules require measurement and periodic reporting of the radioactivity released from nuclear power plants. Gross beta-gamma measurements of unidentified isotopes have been permitted if the concentrations in water were less than 10^{-7} $\mu\text{Ci/ml}$. When isotopic analysis is made to identify specific radioisotopes, higher limits for the individual isotopes are permitted. Most licensees have measured and reported gross activity levels in the past. Increased attention has been given in the last several years, however, to identifying and measuring the amount of radioisotopes released to the environment from operating nuclear power plants. The AEC also conducts independent measurements as an audit, on a spot check basis, of licensee programs. The AEC has co-operated with the Public Health Service (PHS), and later and currently with the Environmental Protection Agency to make special studies of radioactivity around several nuclear power plants [9], and both the AEC and the PHS have worked with the States in developing environmental monitoring programs.

To improve the uniformity and completeness of the monitoring and reporting of radioactive effluents by licensees, the AEC plans to issue safety guides on these subjects in the near future.

2.5. Emergency plans

Although it is judged that the probability of severe accidents is small and the calculated potential consequences of design basis accidents are required to be within the guidelines of the AEC, there remains a small, but non-zero,

probability that significant releases of radioactivity could occur. Since a number of mitigating actions could be taken after such an event to minimize the adverse consequences, the AEC requires that emergency plans be developed at each licensed facility for coping with the consequences of possible accidents, including those in range of severity of the design basis loss-of-coolant accident.

The minimum requirements for emergency plans to be described in applications for construction permits and operating licences are set forth in Appendix E to Part 50 of the AEC's rules, published in December 1970. Also published at this time was a Guide to the Preparation of Emergency Plans for Production and Utilization Facilities to assist applicants in the development of such plans.

Local government agencies and civil authorities, e.g. police, firemen, civil defense personnel, are also involved in planning for response to possible radiation incidents. At the federal level, organization and planning for emergencies has included the development of an interagency radiological assistance plan, supported by regional emergency assistance teams that are prepared to respond immediately to requests for assistance with trained manpower and specialized equipment.

2.6. Nuclear power plant staffing

Although differences in organization and company practices are to be expected and although the AEC administers a program for the licensing of individual reactor operators and senior operators, there has been a need for guidance on the general level of acceptability of the qualifications and training of nuclear power plant personnel. For several years a subcommittee (ANS-3) of the American Nuclear Society Standards Committee worked to develop a standard on these matters, which was approved by the American National Standards Institute in March 1971 as N-18.1-1971, Standard for Selection and Training of Personnel for Nuclear Power Plants.

3. EXPERIENCE WITH THE CONSTRUCTION AND OPERATION OF LICENSED NUCLEAR REACTORS

3.1. General

As of March 31, 1971, 20 nuclear power plants, with a combined capacity of 8121 MW(e), were licensed for operation in the United States. An additional 55 plants (46 221 MW(e)) had been granted construction permits and were being constructed. Applications for construction permits for 25 plants had been filed and were under review by the AEC. Licensed nuclear power plants in operation had accumulated approximately 107 reactor-years of operating experience.

Since the beginning of the 'civilian program' in 1954, the AEC has licensed the operation of 125 power, test and research reactors. Some of them were developmental power reactors, and have since been decommissioned; a few others also have ceased operation. The operation of these facilities has been without a radiation fatality and no instance is known where the operation of a licensed nuclear reactor has resulted in over-exposures to the public or in release of radioactivity off-site exceeding annual limitations set by AEC regulations.

TABLE I. RESEARCH, POWER AND OTHER REACTOR SYSTEMS MOST FREQUENTLY AFFECTED BY UNUSUAL OCCURRENCES^a

	1968	1969	1970	Total
Number of licensed plants in operation	88	97	95	
A. Reactivity control ^b	61	73	41	175
B. Primary system	49	45	47	141
C. Miscellaneous or combination	40	37	28	105
D. Core and internals	34	33	13	80
E. Auxiliary equipment	25	21	22	68
F. Power conversion system	16	9	8	33
G. Electrical equipment	7	17	8	32
H. Emergency power	8	17	5	30
I. Engineered safety features (except ECCS)	8	6	16	30
J. Emergency core cooling	5	2	17	24
K. Containment	2	9	10	21
L. Fuel handling	7	10	4	21
M. Radioactive waste handling	6	8	7	21
Totals	268	287	226	781

^a Instrument malfunctions are included under the affected systems.

^b Includes control rod drive malfunctions.

3.2. Operating experience

Although the record attests to the safe performance of reactor facilities in the United States, abnormal events of many types have occurred and do occur at these facilities. To provide a mechanism for dissemination of information concerning such abnormal occurrences, the AEC publishes 'Reactor Operating Experience' (ROE) reports³. These reports are currently distributed to approximately 1500 individual subscribers around the world.

Because the incidents reported under the ROE programs are representative, rather than all-inclusive, for this paper we have reviewed all unusual occurrence reports for operating reactors made to the AEC over the period 1968 through 1970. The significant results of this review are summarized in Tables I and II.

³ Individual ROE and Reactor Construction Experience (RCE) reports may be obtained by writing to the Director, Division of Reactor Licensing, or Division of Compliance, respectively, USAEC, Washington, D.C. 20545. Compilations of ROE's issued during the periods 1963 to 1965 and 1966 to 1968 have been prepared by the Nuclear Safety Information Center (NSIC), at Oak Ridge National Laboratory, US Department of Commerce, Springfield, Virginia 22151, USA, as NSIC 17 and NSIC 64, respectively. Future updating compilations are planned at approximately three-year intervals.

TABLE II. MOST FREQUENT CAUSES OF UNUSUAL OCCURRENCES (RESEARCH, POWER AND OTHER REACTORS)

	1968	1969	1970	Total
A. Equipment failure	110	145	102	357
B. Component failure	43	55	50	148
C. Other (e.g. fatigue, flow blockage, unidentified)	51	51	33	135
D. Operator error	35	34	33	102
E. Improper design	22	21	30	83
F. Inadequate procedures	11	28	15	64
G. Improper installation	15	12	17	44
H. Improper maintenance	14	11	12	37
I. Improper fabrication	8	7	11	36
Totals	319	347	303	969

Table I is a tabulation of research, power and other reactor systems in which unusual occurrences appeared most frequently. This table identifies those systems for which additional emphasis should be given during design and design review, which require more care in fabrication, and more diligence in maintenance.

Table II is a tabulation of the causes of unusual occurrences most frequently identified in the incident reports. In this table equipment failures include failures of major pieces of hardware, such as diesel generators, control rod drives, or safety valves. Component failures refer to the failures of components within a piece of equipment, such as a relay or solenoid valve within a control rod drive system. Operator errors include such items as manipulation of the wrong valves, failure to follow procedures, and in one case accidental actuation of a control power switch while dusting the console. The improper design category includes such items as the design conditions at the facility which led to a fire in the cable trays as a result of being overfilled and thermally overloaded. Maintenance errors include such occurrences as inadequate lubrication of equipment and maladjustment of switches.

The data represent selected results of a review of approximately 800 reports on file at the NSIC, and were compiled by counting the number of 'keyword' references in the abstracts prepared by the NSIC staff. It should be noted that these data include reports of all types of unusual occurrences. The vast majority of the events were trivial in nature. None involved an actual threat to the health and safety of the general public.

Many of the incidents are reported more than once; for example, the same incident may be reported first as an unusual occurrence and subsequently be covered also in a routine operating report. Such multiple reports, coupled with the fact that most events have more than one contributing cause, preclude a direct correlation between the number of items reviewed and the numbers appearing in the tables.

TABLE III. CONSTRUCTION DEFICIENCY ITEMS - 1970

	Number of items ^a
An important structure, system or component deviated significantly from the material, fabrication or performance specifications with regard to:	
A. A commitment made by the licensee in the application	33
B. AEC Quality assurance criteria (10 CFR, Part 50)	13
C. An industry code or standard	24
D. Industry 'good practice'	7
E. A requirement established by the licensee or his contractors to meet the AEC requirements	7
A required written procedure did not exist, or did not provide for adequate factory or site acceptance, installation, or pre-operational test of an important structure, system or component	2
Records required to demonstrate conformance to AEC requirements were not available, or were incomplete	3
The quality assurance program was not otherwise functioning adequately	6

^a An average of 70 reactors were under construction during the year.

A similar program covering instances of unusual experiences during construction of reactor facilities recently has been instituted. These Reactor Construction Experiences⁴ describe problems encountered in the construction period which may have applicability to other projects.

3.3. Construction experience

For reactors under construction, the periodic inspections conducted by the AEC provide an evaluation of the licensee's Quality Assurance Program and its effectiveness. These inspections involve selected portions of the contractor's Quality Control records, the applicant's Quality Assurance program and a physical inspection of the work in progress. It is the intent of such inspections to verify that the procedures used, the equipment supplied, and the construction performed meet the requirements with respect to commitments made by the licensee in the application, that AEC requirements are met, that industry codes and standards are followed, that industry 'good practices' are followed, and that requirements established by the licensee or his contractors to meet AEC requirements are fulfilled.

A check is made to determine whether all required written procedures are available. Such procedures provide for factory or site acceptance, installation, pre-operational testing of important structures, systems or components, loading of reactor fuel, subsequent low power test operation, or proper handling of emergencies.

⁴ See footnote 3.

A review of records is made to determine whether there is conformance to AEC requirements and whether the records are available and complete.

Table III lists the number of items that were identified as deficient during inspections of the reactors under construction during 1970. These deficiencies were found in about 12% of the more than 200 inspections made at about 70 facilities.

Some deficiencies required substantial effort by the regulatory staff and by the permittees to determine the nature and cause of the deficiency and acceptable corrective action. A few representative examples from past years are as follows.

3.3.1. Cable tray design

A small fire occurred in an electrical cable tray at one reactor. As a result, increased emphasis on the importance of separating the wiring of redundant safety systems has led to careful examination of the ways in which electrical systems are installed on nuclear power plants, and a safety guide on this matter is in preparation. Identification of instances in which wiring for presumably separate and independent safety systems were installed so as to be susceptible to damage from a single incident has led some architect-engineers to devise computerized methods for routing cables and monitoring cable tray capacities.

3.3.2. Piping design

Recent discovery of an instance where major movement of large steam pipes occurred during the course of a severe but not unusual plant transient has led to an investigation of the methods used to design piping and piping support systems and to calculate movements and stresses. It was found that in some instances the design requirements that were being used and that had been developed for conventional plants needed supplementation to avoid unacceptable stress in nuclear power plants.

3.3.3. Cracks in concrete

Severe cracking in the concrete of the dome of a pre-stressed concrete reactor containment building occurred in one case. Investigation was made of the cause and criteria were established for the repair of the dome and for pre-operational and in-service testing programs to assure that the repairs were satisfactory.

3.3.4. Primary system leak

Discovery, during a pre-operational hydrostatic test, of a very small leak in a weld between a control rod stub tube and the pressure vessel led to an investigation of the fabrication techniques for this and other vessels. Many months were required to rectify the design and procedural inadequacies revealed by this event.

3.4. Radioactivity in reactor effluents

A summary of the amounts of radioactivity reported by licensees to be released in effluents from operating nuclear power plants during 1970 is provided in Table IV. The information tabulated was obtained from the periodic reports required from licensees. In addition, the AEC recently has initiated a program of splitting samples with the licensee from the liquid and gaseous waste streams and having the AEC portion analysed at an AEC contractor's laboratory.

3.4.1. Liquid releases

For each of the operating power reactors the gross activity released in effluent waters is listed. Table IV also lists the permissible concentration limits determined by the licensee in accordance with 10 CFR part 20 and the per cent of the limits actually realized on an annual average basis for 1970. The permissible concentration limits determined by the licensee reflect the isotopic identification he has made. If permissible concentration limits had been applied to each individual radionuclide released, the radioactivity released would have represented a substantially smaller percentage of the permissible limit in most cases. The curies of tritium and the per cent of the permissible limit ($3 \times 10^{-3} \mu\text{Ci/ml}$) are also shown.

3.4.2. Gaseous releases

Table IV lists for each of these same plants the curies of noble gases and the curies of particulates released in gaseous effluents. The table shows the permissible quantity limits specified in the facility licence and the per cent of that limit actually realized during 1970. This review indicates that the radioactivity in effluent releases from licensed power reactors generally has been low in comparison with the permissible limits.

3.5. Exposure of personnel

Operators of nuclear power plants are required to submit reports of exposures to employees and visitors subject to surveillance by their personnel monitoring programs. Data for 1969⁵ from 13 operating nuclear power plants show an average exposure of 0.22 rem to a total of 5759 persons monitored. Individual doses were distributed as shown in Table V among various exposure ranges. Most of the exposures in the lowest range were to plant visitors and were less than 0.01 rem per person.

In summary, with only 20 nuclear power plants now in operation, including mostly older and smaller types with only a few much larger types, but with more than 55 additional plants in construction ranging up to 1170 MW(e) in size, having high fuel burn-up, higher power densities and greater sophistication and complexities of design, the depth of safety review and of safety requirements in design, construction and operation have had to advance rapidly to keep pace with advancing reactor technology. The principal objective of the AEC's regulatory program will continue to be

⁵ Tabulation of data for 1970 was not yet completed at the time this paper was prepared.

TABLE IV. RELEASES OF RADIOACTIVITY REPORTED BY NUCLEAR POWER REACTOR OPERATORS IN THE UNITED STATES IN 1970^a

Facility	In liquid effluents ^b					In gaseous effluents ^c					
	Mixed fission and corrosion products			Tritium		Noble and activation gases			Halogens and particulates		
	Released curies	Concentration limit (10 ⁻⁷ µCi/ml)	Per cent of limit	Released curies	Per cent of MPC	Curies released	Curies permissible	Percentage of permissible	Curies released	Curies permissible	Percentage of permissible
Indian Point 1	7.8	70	0.28	410	0.03	1 700	5.3 × 10 ⁶	0.03	7.5 × 10 ⁻²	7.5	1
Yankee Rowe	0.034	1	0.14	1500	0.21	17	6600	0.26	1.8 × 10 ⁻⁶	0.44	4.1 × 10 ⁻⁴
San Onofre	3.8	1	6.3	4800	0.26	1 500	5.7 × 10 ⁵	0.28	None detected		
Conn Yankee	22	9	3.7	7400	0.36	700	2.8 × 10 ⁵	2 × 10 ⁻³	4.6 × 10 ⁻⁴	0.3	0.22
Guima	9.3	3	4.5	110	0.005	10 000	3.6 × 10 ⁵	2.8	None detected		
H.B. Robinson 2	0.12	1	4.1	2.7	0.003	0.012	4.7 × 10 ⁴	2.6 × 10 ⁻⁵	None detected		
Saxton	0.26	13	3	10	0.05	2 200	3.7 × 10 ³	58	0.025	10	0.25
Oyster Creek	4.5	1	17	22	0.0028	110 000	9.4 × 10 ⁶	1.2	0.31	130	0.25
Nine Mile Point	9.4	1	21	20	0.0015	9 500	27 × 10 ⁶	3.7 × 10 ⁻²	< 0.06	63	< 0.1
Dresden 1	8.2	1	25	5	0.0005	910 000	18 × 10 ⁶	5.2	3.3	75	4.3
Dresden 2	13	1	15	31	0.0012	280 000	22 × 10 ⁶	1.3	1.6	110	1.4
Humboldt Bay	2.4	1	15	<7	<0.0014	5.4 × 10 ⁵	16 × 10 ⁵	34	0.35	5.7	6.17
Big Rock Point	4.7	15	3.1	53	0.018	280 000	3 × 10 ⁷	0.88	0.13	37	0.34
LACBWR	6.4	300	0.067	20	0.002	700	3.2 × 10 ³	0.27	<0.063	16	<4
Fermi 1	0.04	1	0.54	None detected		<93	1 × 10 ³	<8.4	<1 × 10 ⁻³	7 × 10 ⁻³	<14
SEFOR	4 × 10 ⁻⁶	1	5.2	3.7 × 10 ⁻³	0.16	0.06	25 × 10 ³	2.4 × 10 ⁻⁴	1.8 × 10 ⁻⁶	0.17	0.001
Peach Bottom	6 × 10 ⁻³	1	0.15	< 50	0.043	5.7	19 × 10 ⁴	3 × 10 ⁻³	< 6 × 10 ⁻⁴	0.1	<0.6

a From licensees' reports.

b Facility licences require that the release of radioactive liquids in plant effluents be in accordance with 10 CFR Part 20, 'Standards for Protection Against Radiation'. For mixtures of radionuclides in the effluent, Part 20 provides two alternatives for determining permissible concentration limits. If the identity and concentration of each nuclide is known, Appendix B, Note 1, prescribes a formula for calculating the limiting value. Note 3 prescribes a method for selecting one of a series of values if it can be shown that certain radionuclides are not present in the mixture. The values calculated or selected by licensees may vary from year to year.

One of the limits specifically mentioned in Note 3.c. of Part 20 is 1×10^{-7} $\mu\text{Ci/ml}$, which is sufficiently restrictive that it can be used for any mixture of fission and corrosion products in water from any nuclear power reactor without any identification of the radioisotopic composition of the mixture. Typical isotopic compositions of radioactivity in water from power reactors are such that limits higher by two orders of magnitude or more are expected to be available to the licensee if he wishes to support them with adequate radioisotopic analyses. The per cent of limit given in this column generally represents upper bounds to the value that would be applicable on the basis of a complete analysis of the composition.

c Where the technical specifications express a release limit in terms of a constant factor times the 10 CFR Part 20, concentration limits, the MPC used is 3×10^{-8} $\mu\text{Ci/cm}^3$. This MPC is based on typical noble mixture releases with less than two hours holdup. (For a holdup longer than two hours the MPC is larger.)

Where the technical specifications do not state an annual limit for the iodines and particulates, values of 1×10^{-10} $\mu\text{Ci/cm}^3$ and 3×10^{-11} $\mu\text{Ci/cm}^3$, respectively, were used. These MPC's are based on the most restrictive isotopes normally found — ^{131}I and ^{90}Sr . The annual limit was reduced by a factor of 700 to account for reconcentration.

TABLE V. PERSONNEL EXPOSURES — 1969

Range in rems	No. of persons
0 - 1.25	5352
1.25 - 2	160
2-3	144
3-4	70
4-5	26
5-6	5
6-7	2
More than 7	0

focused on prevention of serious accidental releases of fission products to public areas. Thus far, no serious consequences of radioactive releases from nuclear plants have occurred, but the number and nature of some of the many abnormal events which have been experienced give no basis for relaxation of vigilance. Neither a stable level of standardization in reactor design nor a fully adequate set of reactor safety requirements now exists.

In addition, a general world-wide upsurge of public interest in environmental protection has broadened the scope of regulatory concerns to include re-examination of standards for effluents, greater control of waste handling and management and extension of attention to thermal and other environmental effects in addition to radiological concerns.

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REACTOR SAFETY IN THE FEDERAL REPUBLIC OF GERMANY

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Abstract—Résumé—Аннотация—Resumen

REACTOR SAFETY IN THE FEDERAL REPUBLIC OF GERMANY.

Licensing practices in the Federal Republic are most advanced for light-water reactors. Safeguards are primarily for the control of design basis accidents. For the definition of such accidents, reliability considerations play an important part. Advantages and limits of the basis accident concept and the probability concept are discussed and illustrated by some examples. The permissible maximum risk for the individual is discussed and compared with the actual exposure in certain incidents.

Licensing of light-water reactors has been standardized to a great extent and has led to safety criteria, many of which apply to components of the pressure system. Other items are the design and fabrication of the containment interior, the redundancy and diversity of the emergency cooling system, the design and testing of the containment and penetrations, the internal ventilation, the post accident instrumentation, the pressure suppression system, etc. For high-temperature reactors and fast reactors, the criteria are still under development. The requirements for repeated inspections and tests of the plant are explained and actual incidents of relevance to safety are described.

A research program provides additional information and some results are reported. They include results from experiments on a sub-assembly scale on the capacity of the emergency cooling and burn-out behaviour. Another large-scale test is for the flow condition in the vessel interior and the flow rate during a basic accident. Others are for the development of remotely controlled ultra-sonic testing equipment, sound detection during fracture growth and fracture mechanics of primary components. The programs for high temperature reactors concern the concrete vessel, and those for fast reactors cover propagation of local cooling disturbances.

BASF has, for the first time, applied for licensing of an in-city site for a nuclear power station. A survey of population conditions at licensed sites is given. Special considerations for urban sites are discussed.

LA SECURITE DES REACTEURS EN REPUBLIQUE FEDERALE D'ALLEMAGNE.

La pratique la plus développée pour obtenir l'autorisation de construire un réacteur nucléaire en République fédérale d'Allemagne est celle suivie pour les réacteurs à eau légère. Les garanties concernent notamment le contrôle des accidents de base. Les considérations de fiabilité y jouent un rôle important. Les avantages et les limites des accidents de base aussi bien que la conception de la probabilité sont discutés et expliqués à l'aide de quelques exemples. Le risque maximal admissible pour les personnes est discuté et comparé à l'exposition réelle au cours de quelques incidents.

La procédure d'autorisation a été standardisée dans une large mesure et, par conséquent, des critères de sécurité ont été mis au point dont un grand nombre s'appliquent aux composants du système de pression. D'autres concernent la conception et la construction de l'intérieur de l'enceinte étanche, la redondance et la diversité du système de refroidissement de secours, la conception et l'essai de l'enceinte étanche et des traversées, la ventilation interne, les appareils de mesure installés après un accident, le système de suppression de la pression, etc. En ce qui concerne les réacteurs à haute température et les réacteurs à neutrons rapides, les critères sont en cours de développement. La nécessité de soumettre l'installation à des inspections et des essais réguliers est expliquée et les incidents qui présentent une importance du point de vue de la sécurité de l'installation sont décrits.

Un programme de recherche fournira des renseignements supplémentaires. Quelques résultats sont indiqués. Ils comprennent des expériences à l'échelle du sous-assemblage qui portent sur la capacité du système de refroidissement de secours et le comportement à la combustion. Un autre essai exécuté à grande échelle concerne les conditions du flux existant à l'intérieur de la cuve et le taux de fuite pendant un accident de base. D'autres essais portent sur la mise au point d'appareils d'essai ultrasoniques télécommandés, la détection du son durant la fissuration et la mécanique des ruptures sur les composants du circuit primaire. Les programmes relatifs aux réacteurs à haute température se rapportent au caisson en béton, tandis que les programmes portant sur les réacteurs à neutrons rapides s'intéressent à la propagation des perturbations locales de refroidissement.

Pour la première fois la BASF a demandé l'autorisation de construire une centrale nucléaire en ville. La situation démographique aux environs des emplacements autorisés est décrite. Les considérations particulières aux emplacements situés dans des centres urbains sont discutées.

БЕЗОПАСНОСТЬ РЕАКТОРОВ В ФЕДЕРАТИВНОЙ РЕСПУБЛИКЕ ГЕРМАНИИ.

Процедуры лицензирования в Федеративной Республике Германии лучше всего разработаны для легководных реакторов. Нормы безопасности установлены, главным образом, применительно к основным авариям. При определении этих норм важную роль играют соображения надежности. На ряде примеров в докладе рассматриваются и объясняются преимущества и недостатки понятий основной аварии и ее вероятности. В докладе рассматривается максимально возможная опасность для человека и дается сравнение с фактическими дозами облучения при отдельных конкретных авариях.

Лицензирование легководных реакторов в значительной степени стандартизовано, это дало возможность выработать критерии безопасности. Многие из этих критериев касаются компонентов систем для поддержания давления. Другие применяются к конструкции и изготовлению внутренних компонентов противоаварийной оболочки реактора, надежности и многократному дублированию аварийных систем охлаждения, конструкции и испытанию оболочки реактора и системы трубопроводов, системе внутренней вентиляции, аварийной контрольно-измерительной аппаратуре, системе гашения давления и т.д. Для высокотемпературных реакторов и реакторов на быстрых нейтронах такие критерии еще разрабатываются. В докладе разъясняются требования к периодическим инспекциям и испытаниям установок. Дается описание фактических аварий с точки зрения безопасности.

В докладе приводится дополнительная информация о программе исследований. Сообщается о некоторых результатах, в том числе о результатах экспериментов с критическими сборками, проводимых с целью изучения мощности аварийных систем охлаждения и режима выгорания топлива. Другой крупный эксперимент связан с изучением режима потока в пределах бака реактора и скорости потока во время основной аварии. К числу других исследований относится разработка опытного оборудования на ультразвуке для дистанционного управления, обнаружение с помощью звука нарушений компонентов первого контура и изучение их механики. Программы для высокотемпературных реакторов предусматривают проведение экспериментов с реакторными корпусами из бетона, а программы для реакторов на быстрых нейтронах включают изучение вопросов распространения местных повреждений в системе охлаждения.

Компания "BASF" впервые обратилась с просьбой о выдаче разрешения на приобретение площадки под строительство атомной электростанции в черте города. В докладе приводятся данные о населении в районах размещения станций, разрешенных для строительства. Особо рассматриваются вопросы, связанные с размещением станций в городских районах.

SEGURIDAD DE LOS REACTORES EN LA REPUBLICA FEDERAL DE ALEMANIA.

En la República Federal Alemana, la reglamentación para la concesión de licencias muestra los mayores progresos en el caso de los reactores de agua ligera. Las garantías de seguridad se refieren principalmente al control de los accidentes básicos. Para su definición, las consideraciones de confiabilidad representan un papel importante. En este trabajo se analizan y aclaran por medio de algunos

ejemplos las ventajas y límites de los conceptos de accidente básico y de probabilidad. Se define el riesgo máximo permisible para los individuos y se compara con la exposición real en ciertos accidentes.

La concesión de licencias para reactores de agua ligera se ha normalizado en gran parte y ha llevado a ciertos criterios de seguridad, muchos de los cuales se aplican a los componentes del sistema de presión. Otros aspectos son el diseño y fabricación de la contención interior, la duplicación y diversidad del sistema de refrigeración de emergencia, el diseño y comprobación de la contención y penetraciones, la ventilación interna, la instrumentación post-accidente, el sistema de supresión de presión, etc. Los criterios para los reactores de alta temperatura y para los reactores rápidos están todavía en estudio. En la memoria se explican las condiciones que exigen inspecciones y pruebas repetidas de la planta, y se describen sucesos reales de relevancia para el control de la seguridad.

Un programa de investigación suministra información adicional y se informa sobre algunos de sus resultados. Estos incluyen algunos resultados de experimentos a escala de subconjunto sobre la capacidad de la refrigeración de emergencia y el comportamiento del abrasamiento. Otra prueba a gran escala se relaciona con la condición del flujo en el interior de la vasija y el caudal de flujo durante un accidente básico. Otros resultados conciernen al desarrollo del equipo ultrasónico de comprobación controlado a larga distancia, la sonda-detección durante el crecimiento de la fractura, y las características mecánicas de la fractura de componentes primarios. Los programas para reactores de alta temperatura a la vasija de hormigón, los de los reactores rápidos incluyen la propagación de las perturbaciones en la refrigeración local.

El BASF ha solicitado por primera vez el permiso para el emplazamiento urbano de una central nuclear. La memoria contiene un informe de las condiciones demográficas en los lugares en que se ha concedido permisos. Se tratan las especiales consideraciones para los emplazamientos urbanos.

1. GENERAL

Nuclear installations in the Federal Republic of Germany are licensed on the basis of the Atomic Energy Act of 23 December, 1959. The licence is granted by the supreme authorities of the States (Länder), to whom the Federal Minister for Education and Science is entitled to give directives. In general, the licensing authorities are advised by the Technische Überwachungs-Vereine (TÜV)¹. These private organizations have been in existence for about one hundred years and also control compliance with safety regulations in many other fields. Furthermore, the TÜV supervise the manufacturing of components and the construction and operation of the installation.

In particular, the TÜV initiated the Institut für Reaktorsicherheit (Institute for Reactor Safety, IRS), which conducts basic and special studies for the Federal Minister for Education and Science and carries out safety reviews of nuclear installations. The Federal Minister for Education and Science is advised by the Reactor Safety Committee (RSK), which consists of about twenty independent specialists from various disciplines.

Altogether, the safety review of a nuclear power plant and its inspection during the construction period take about fifty man years of work by TÜV, IRS and RSK. Up to now, seventeen research reactors, nineteen nuclear power plants (mainly with pressurized-water or boiling-water reactors) and seven other nuclear installations (fuel element manufacturing, reprocessing, etc.) have been completely or partially licensed.

¹ Technical Supervision Associations (also referred to as Inspectors' Organizations).

2. TECHNICAL AND SCIENTIFIC FUNDAMENTALS: THE DESIGN BASIS ACCIDENT

The basis for the safety evaluation of a reactor installation is the concept of the design basis accident. As is generally known, this is an extreme accident, which is fundamental to the design of the engineered safeguards. This concept, of course, also includes studies of different causes of accident and courses to determine the need for and the design of certain engineered safeguards.

The preliminary hazard study is used to investigate dangerous conditions which can be caused by energy releases from potentially existing sources within the installation.

By means of the failure mode and fault effect analysis, all possible modes of component failure and their immediate consequences are studied. This is a suitable tool for carrying out a fault tree analysis. The method seeks out all causes which can initiate a certain undesired event. The fault tree demonstrates dangerous failure combinations which lead to an accident.

The accident course analysis investigates the entire chain of events rather than the immediate consequences of only component failures, as is done by the fault effect analysis.

The concept of the design basis accident has been compared with the probability concept of reactor safety as proposed, for example, by Farmer [1]. In a previous paper [2], the authors treated Farmer's model, the techniques of reliability analysis on which it is based and its application in the Federal Republic. They considered it a valuable step towards an objective risk evaluation, which should be further developed with all available effort. However, for the time being the general application of the probability concept meets with a few obstacles which are partly of principle and partly of a practical nature.

The application of the probability model requires the fulfilment of a few important conditions:

- (a) there must not be any unaccounted causal interrelations between different failures;
- (b) all components must be free from severe, statistically unaccountable systematic faults in construction, manufacturing and operating conditions, that is, only type-tested components may be employed;
- (c) human error, if not included in the analysis, must not be a possible cause of damage;
- (d) the temporal distribution of the failure probability or the failure rate, respectively, must be known.

Experience of how to guarantee the fulfilment of these conditions completely and in each case is still lacking to a large extent. This is particularly true for points (b) and (c), the systematic faults and human error.

Practically no statistical data are available on the failure probability of certain components such as pressure vessels.

For very severe but very rare hypothetical accidents, the main difficulties are for a treatment which is based entirely on probability considerations. The rupture of a primary coolant pipe of a pressurized-water reactor, together with the simultaneous failure of the emergency core

cooling system, which under certain circumstances can cause damage to the containment through a hydrogen-oxygen explosion (zirconium-water reaction), is assumed by several authors to occur with a frequency of less than $10^{-7}/\text{yr}$. Thus, even with all the installations in operation in the world, the law of large numbers is not meaningful within human life expectancy. These probabilities are therefore meaningless for quantitative assessment, which, in this respect, would be no better than an artificially defined design basis accident. All these arguments are not directed against the application of the probability concept in safety analyses. They are, however, directed against its exclusive and overly schematic use.

The concept of the design basis accident, as we use it, does not by any means exclude the application of certain probability techniques but puts them into a certain relationship. The term "design basis accident" obviously includes an essential element of definition and convention, which should serve as unequivocal design criteria for the manufacturer. The basis of the definition of this one particular accident as the basis for design follows from interrelations between conservative assumptions, previous engineering practice, considerations for the optimization of safeguards under the aspect of operation and, finally, from certain probability considerations. This is demonstrated in greater detail for the particular cases of pressurized and boiling-water reactors.

The design basis accident for the pressurized-water reactor consists of the instantaneous double-ended failure of a primary coolant pipe which leads to the opening up of a cross-section of $2 \times \text{ca. } 4800 \text{ cm}^2$. We also include in the concept of the accident the possibility of a corresponding rupture cross-section in the pressure vessel proper, immediately at the nozzle. For the rest of the pressure vessel, particularly for the lower part, considerations based on fracture mechanics lead to the assumption of maximal fracture sizes up to, but not including, the critical crack length (approximately 850 cm^2) for the design of engineered safeguards. The design covering other accidents is treated later in the paper.

For the boiling-water reactor, the design basis accident is also essentially a definition. It is taken as the double-end failure of the largest coolant pipe (recirculation line) with a blowdown cross-section of $2 \times 2237 \text{ cm}^2$. In comparison with the state of development which has been achieved in the Würgassen nuclear power plant, the newer boiling-water reactors use internal axial pumps for coolant recirculation, which make the recirculation loops superfluous. The pipe cross-section with the highest blowdown rate (feedwater pipe) is considerably smaller than that of the recirculation pipe. Nevertheless, in the sense of the concept discussed here, the hypothetical rupture cross-section of the former magnitude is still used. For the pressure vessel the same considerations are valid as described above for the pressurized-water reactor.

3. DEFINITION OF RISK LIMITS

In addition to the problem of determining failure probabilities of components and systems or, in more general terms, of the risk which is connected with nuclear power generation, one has to raise the question of the magnitude and the acceptability of these risks. So far, no competent institution has expressed an opinion on this problem in the form of

recommendations or regulations. All proposals for an evaluation of risks on the basis of probability which have been made by different authors in the Federal Republic of Germany should be understood as contributions to the preparation of a concept for the consequent and quantitative assessment of, if possible, all reactor safety problems. A decision on the tolerable risk limits will have to be made by an authorized body at a later time and include more than technical-scientific considerations. In the following text, one possibility of approaching this problem is discussed. It was described in detail in a paper by Lindackers [3] and in a monograph [4] published in 1970.

As a consequence of an accident j , which has the frequency H_j (yr^{-1}), an individual receives a dose D_{ij} (rem) to the body organ i after an activity release. By multiplying the probability of occurrence H_j with the received dose D_{ij} for this particular accident, a mean annual radiological burden can be defined, whereby the total mean annual burden is obtained by a summation over all single exposures caused by the different accidents j .

As an example, let us consider the mean whole-body dose for the design basis accident. The frequency of occurrence of the DBA can be assumed to be less than $10^{-4}/\text{yr}$, for the pressurized as well as the boiling water reactor. The whole-body dose received in the environment as a consequence of this accident is in the neighbourhood of 1 rem, if pessimistic assumptions for fission product release and dispersion are made. The mean annual radiological burden, which is calculated under these assumptions, amounts to less than 0.1 mrem and has to be compared, for example, with the natural irradiation of about 150 mrem/yr. This comparison shows clearly the very low hazard potential of the design basis accident quoted above. Furthermore, the mean annual radiological burden of the design basis accident is more than one order of magnitude less than the radiological burden received as a consequence of normal operation, which has to be assumed as 1 mrem/yr.

The indexed damage can be divided into three groups: late somatic effects, genetic effects and radiation deaths.

(a) Late somatic effects²

The late effects can be related to the received dose by means of a risk coefficient K_i (cases/persons $\times$ rem), which has to be determined for each possible resulting type of damage. Then, with the assumption that no threshold dose exists,

$$K_i D_{i, \text{mean}} = \sum_j K_i D_{ij} H_j \leq R_i^g \left(\frac{\text{cases}}{\text{person} \times \text{yr}} \right)$$

where R_i^g is a risk limit which must not exceed a fraction of the natural risk of contracting these diseases. For evaluating the total risk, the radiological burden caused by normal operation has to be added to the mean annual radiological burden. Indications for obtaining the numerical values of K_i and R_i^g can be found in the literature [5].

² "Late somatic effects", in the sense used here, are radiation damage to organs which becomes manifest only after a long latent period, i.e. after years or decades.

(b) Genetic effects

The genetic radiation risk can also be treated on the basis of the concept of the mean annual radiological burden. Since the genetic risk consists primarily of the generation of recessive mutations, the genetic dose to a population has to be considered for evaluation of the genetic effects rather than the radiological burden of the individual. To estimate the genetic risk, which is caused by an additional radiological burden from the application of nuclear energy, the natural irradiation can be used for comparisons.

(c) Lethal radiation doses

If, in an accident analysis, one assumes the possibility of failure of engineered safeguards, the risk of humans receiving lethal radiation doses must also be considered. This risk should be compared with other risks of dying from accidents in comparable conventional plants (cf. Ref. [6]).

In evaluating risks, the fact that several reactors may contribute to the total risk if they are constructed in a narrow and limited area must be taken into account. It should finally be noted that the evaluation of risk limits as described would have to be extended by further consideration if a larger number of victims were expected as a consequence of the same accident. This shows very clearly that a decision on tolerable risk limits is essentially of a political nature.

4. SAFETY CRITERIA

Experience acquired so far with safety analyses and evaluations in the course of licensing procedures for nuclear power plants have led to the definition of safety criteria which are now applied to the design and the safety review of new projects with thermal reactors [7]. These criteria, which were drafted by the Institute for Reactor Safety, are based on the concept of the design basis accident. They include

- (a) requirements for the design, manufacturing and inspection of components;
- (b) regulations for the design and reliability of engineered safeguards for the limitation of the consequences of accidents.

The general character of the safety criteria, which was thought to be advantageous when the criteria were formulated, did not allow the special features of the various reactor types to be dealt with in detail. A first and necessary addition to the criteria has been made in the meantime in the form of two documents [8, 9], drafted by the Reactor Safety Committee, which are concerned, in particular, with nuclear power plants with boiling- or pressurized-water reactors as offered by German manufacturers.

These RSK guidelines include provisions for the design of pressure suppression systems, which are usual with these boiling-water reactors, for containment with a re-vent system, for secondary containment and for a rod ejection grid. For the pressurized-water reactors, which are surrounded by a full pressure steel containment vessel and an additional concrete shell as a secondary containment, provisions are made, for

instance, for containment isolation, the enclosure of containment penetrations in chambers, and for the design of the full pressure containment.

It might be of interest here to indicate a few differences between our safety criteria and the USAEC design criteria published in 1967.

The IRS criteria mentioned above include definitions of the various operating and accident conditions and descriptions of types of damage which are characteristic and can be considered acceptable for the various cases. Furthermore, the unconditional requirement of certain safeguards such as a containment vessel or special emergency core cooling systems has been avoided. To account for the special characteristics of reactors other than water-moderated reactors, particularly of gas-cooled reactors, the criteria state the conditions under which such safeguards have to be provided. Gas-cooled reactors of the THTR or AGR type do not, in our opinion, need an additional containment vessel. As examples of other differences it might be mentioned that, in our criteria, one hundred per cent non-destructive testability of the pressure vessel shell is also required in recurring inspections, and that, with a suitable instrumentation and shutdown system, a positive power coefficient would not necessarily be considered unacceptable. On the other hand, our criteria include the requirement to design the containment vessel against the build-up of pressure differentials among the various compartments after a coolant loss accident; they also include detailed provisions for the separation of process instrumentation and safety system and the requirement of a secondary containment for both pressurized- and boiling-water reactors.

According to our requirements for the redundancy of emergency core cooling systems, at least $3 \times 100\%$ or $4 \times 50\%$ of the required cooling capacity has to be provided (Fig. 1). This requirement, which also applies to the redundancy of the emergency power supply systems, is based on the consideration that, as an immediate consequence of the design basis accident, one emergency cooling system becomes ineffective. The remaining systems have to be so designed that they will supply adequate long-term cooling even if a further system fails.

Another difference is the fact that we do not accept "safe ends" - pipe connections inside the biological shield, which by definition cannot fail. The rupture of a pipe is, in our accident analysis, considered possible at each point; precautions are also required against pressure vessel damage up to a certain level.

Quantitative requirements for the reliability of engineered safeguards are made on the basis of relative standards. The higher the required effectiveness, the more reliably a safeguards system must operate. In such a manner, the risk that is the product of the consequences and the probability of damage remain constant as related to the design basis accident.

Safety and supply systems, for which redundancy is required, must also be separated spatially because of the possibility of simultaneous damage to these systems from the same cause.

The basic requirement for the effectiveness of emergency core cooling systems, which are assumed to function in the accident analysis, defines that cladding ruptures or fuel element deformations must not occur in the course of the design basis accident: therefore, the maximum admissible cladding temperatures are $800 - 900^\circ\text{C}$.

The design basis accident has to be analysed with all its consequences being taken into account. The resultant forces during blowdown can occur

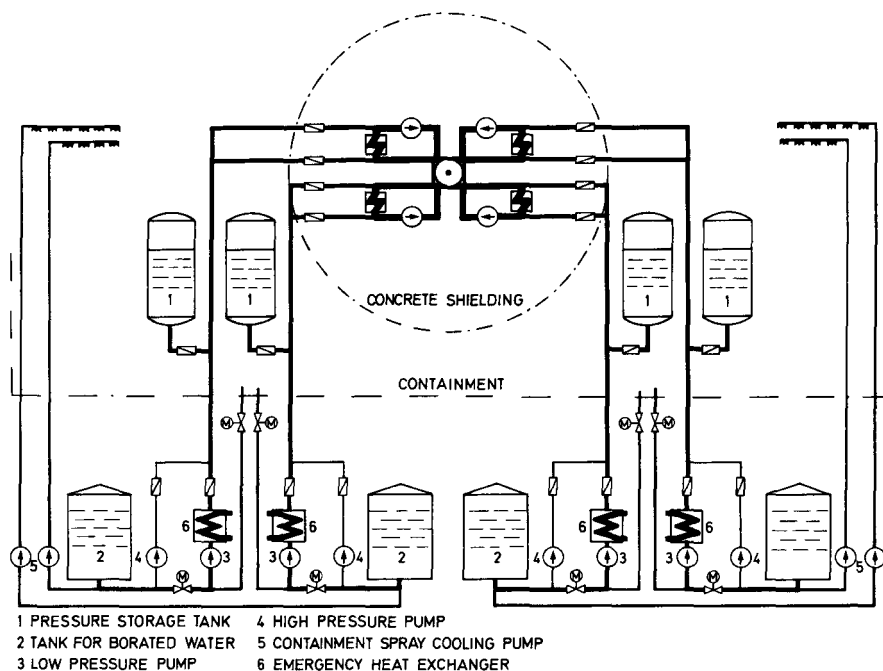


FIG.1. Emergency core-cooling system for a PWR.

in every direction, and the support structures inside and outside the pressure vessel should thus be designed accordingly.

The containment and its internals, particularly the instrumentation, have to be designed in such a way that recurring tests of the leakage rate can be carried out under full accident pressure. Electric motors for emergency circuits are not admissible inside the containment.

Reactors must be designed and constructed such that, during their entire lifetime, recurring inspections can give a clear picture of the safety and availability of the plant. On the basis of these considerations, we published recommendations in 1969 for recurring tests of important parts of nuclear power plants [10], and these recommendations have since been further elaborated.

As indicated above, special criteria for fast reactors do not yet exist. Since the licensing procedures for the first prototype nuclear power plant with a fast reactor in the Federal Republic of Germany are currently in progress, it is now too early to define such criteria. Nevertheless, there do exist a number of requirements which will have to be met by the design of the projected type (sodium-cooled, loop type).

The opinion in the Federal Republic, based on the ICRP recommendations and the Euratom standards, is that the allowable genetic dose of 5 rem/30 years, quoted for the total population, must not be used up exclusively by nuclear technology. In our view, the radiological burden to the population caused by nuclear energy must not exceed 2 rem/30 years, whereby the following additional conditions have to be met:

- (a) one half of this value (1 rem/30 years) is assigned to the burden from airborne radioactivity;
- (b) the other half, also 1 rem/30 years, includes all other discharges, including the burden from ecological chains and drinking water;
- (c) these limits are applicable to the population group living in the area where the highest concentrations of the radioactive effluents occur or experiencing the highest exposure as a result of dietary or other habits;
- (d) permits for the discharge of radioactivity are granted only within these limits and only up to a value which has been demonstrated to be necessary.

In the authors' view, this system guarantees that the additional radiological burden caused by the application of nuclear energy will not, even for the most exposed part of the population, exceed limits corresponding to a few per cent of natural irradiation.

During the last few years, much of the discussion has been on the possibility of external damaging effects to a nuclear power plant. While we believe that natural catastrophes such as floods, earthquakes, etc., can be excluded as causes of large releases of radioactivity if a plant is designed according to corresponding criteria developed for nuclear installations, other potential causes always create new problems. This became particularly evident when, in the BASF project, the use of a nuclear power plant for the generation of process steam within a chemical plant was discussed for the first time. Both here and in a number of other projects, the possibility of damage to the nuclear power plant from the explosion of chemical products or raw materials – for example mixtures of evaporated liquefied gases and air – has been studied in detail. All these studies have shown that it is necessary to design the plants against certain pressure waves that can occur as a result of such explosions and the structures of a number of nuclear power plants have been designed and tested with this in mind. Consideration of the probability of such events – or in other words, a comparison of risks – is the best basis for decisions on whether structural protection is required or not.

Safety criteria for nuclear power plants, even if they are very detailed, can never be perfect. Nevertheless, in the Federal Republic attempts are being made to develop the existing criteria further, to define new ones and to quantify the safety requirements as much as possible. Accordingly, work is in progress at the present time in the following areas:

- (a) more type-specific criteria;
- (b) introduction of numerical values for minimum reliabilities of components and systems;
- (c) criteria for discharge systems for airborne and liquid waste;
- (d) criteria for design against external causes of damage.

At the present stage of development of nuclear power in the Federal Republic of Germany, a nuclear power plant can no longer be looked upon as a single isolated installation. All prognoses indicate that, in this relatively small and densely populated country, there will be about 200 large nuclear power plants within the next two or three decades, concentrated essentially along the larger rivers. Further development of the criteria will have to provide for this future.

5. INCIDENTS

The reports and evaluations of incidents are important for operators, manufacturers and safety assessors. Important conclusions can also be drawn from "almost" accidents. A registration and reporting system in the Federal Republic is still in the preparatory phase, but a few incidents are reported below.

(1) In a water-cooled reactor, a loss of primary coolant from the water purification system caused a pressure build-up in the containment. The cause of the leakage was a defective gasket in the purification system. This defect in turn was caused by the arrangement of the valves in the circuit and their failure, which led to a pressure build-up within one part of the piping, which had not been foreseen at the design stage.

(2) In a boiling-water reactor, the leakage from the stuffing box of a large valve in the forced circulation lines had increased considerably. The primary coolant was released as a steam-water mixture, and caused vibrations in the corresponding leakage-cooling system, which led to a rupture of a cooling coil. Thus, primary coolant flowed through the stuffing box leak and the cooling circuit into the river and contaminated the river water. Although conservative estimates led to the assumption that about 18 Ci of radioactive contaminants had been released together with the primary coolant (maximum admissible annual release: 5.4 Ci), the contamination level was so low that not a single isotope from the primary coolant could be found and identified in the river water.

(3) In another boiling-water reactor, about ten tons of primary water were released into the reactor building when a mechanic inadvertently opened a sealing water filter without having first closed the shutter. The filter was ejected by the sealing water, which was under more than reactor pressure; the mechanic was scalded by the released steam and was therefore unable to close the shutter. Together with his colleagues he left the reactor building.

Since the sealing water supply was interrupted, the corresponding pump – a recirculation pump – failed and caused a reactor shutdown. Attempts were made to maintain the water level by means of different circuits and to isolate the leak. When protective clothing was used, the leak was successfully isolated. In the meantime, the sealing water supply had started operating again and steam generation had decreased. A systematic search for the defect showed that the check valve in a connection between the sealing water supply and the emergency core-cooling system had no valve case. In fact, the sealing water pump had been feeding through the leak caused by the mechanic's neglect while the reactor was operating at full pressure.

(4) Another incident in a large nuclear power plant should be mentioned, which was potentially very dangerous for the reactor pressure vessel. The faulty operation of switches caused water from the fuel storage pool to flow into the reactor well on top of the reactor pressure vessel, where it came in contact for a short time with the flanges of the pressure vessel under operating temperature. When the error with the switches was noticed, the reactor well drainage was immediately opened.

The thermal shock which occurred as a result of the cooling of the lid and vessel flange, as well as the stress of the lid bolts in the course of

reheating after drainage of the reactor well, was considerable in respect to the integrity and leak-tightness of the pressure vessel. The highest stress occurred in the lid bolts, when the lid flange was reheated faster than the bolts after draining the water from the reactor well. During the cooling-off period, the tensile stress of the bolts was below normal, which could possibly have caused a leak in the sealing of the lid. Control calculations for the lid bolts, which had been carried out on the basis of temperature recordings, resulted in values which remained within tolerable limits. After the flooding valve had been connected to the equipment storage pool, which is used during refuelling, another incident of this kind was considered impossible.

The evaluation of these incidents cannot give a representative picture; however, the following statements are admissible:

- (1) most incidents are caused by human error;
- (2) the techniques of systems analysis have consequently to be applied to the planning and assessment of circuits;
- (3) in all cases, the engineered safeguards have operated faultlessly, and an employee or the environment has never been exposed to radiation above the tolerable limit.

6. RESEARCH PROGRAMS

In the course of the past year, research and development in the field of reactor safety were substantially expanded and intensified in the Federal Republic of Germany, particularly on the initiative of the Minister for Education and Science and on the basis of his willingness to finance these projects. The objective of this enlarged research program is the investigation of possible causes of, and effective protective measures against, extreme accidents. A few of the more advanced projects are outlined below.

6.1. Research program on emergency cooling for water-cooled reactors

The primary interest was in better clarification of the thermal-hydraulic conditions prevailing after primary piping ruptures in the reactor core, particularly during the high-pressure phase. A corresponding experimental program is being conducted by AEG.

In various experimental loops, the cooling channels of a reactor core and some components of the primary system can be simulated. Starting with single-tube or single-rod tests, directly and indirectly heated fuel-rod bundles for pressurized- or boiling-water reactors are being investigated at full scale. Values for depressurization rates and flow changes, which have been predicted and verified by large-scale tests, are obtained by using motor-operated valves. In addition, rupture disks are employed to simulate loss-of-coolant accidents. Flow reversals and the influence of large volumes on heat transfer in the cooling channel can be simulated. The heating power corresponds to the maximum bundle or rod power in the reactor. The tests are carried out with both constant and variable axial power distribution. With respect to the significance of the experimental results, great value was placed on suitable fuel-rod simulation and on proper instrumentation.

Parallel with these experiments, the low-pressure phase of the loss-of-coolant accident is being studied by Siemens. The program investigates mainly the distribution of coolant over the reactor cross-section during spraying and flooding, the rate of penetration into the cooling channels and the re-wetting of hot fuel rods. Again, starting from single-rod tests, entire fuel elements for boiling- or pressurized-water reactors are being investigated, and the effect of parallel fuel elements is simulated by testing suitable sections of neighbouring bundles.

In the final phase of the experimental program, integral tests with a simulation of the entire accident and including the conceivable emergency cooling actions will be conducted. The AEG test equipment is being expanded accordingly.

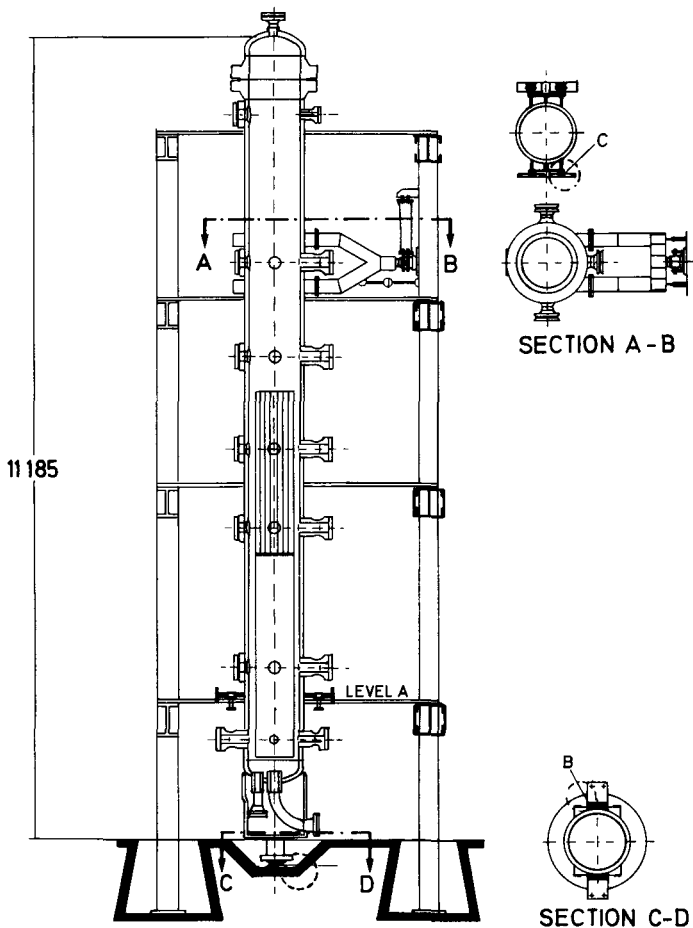


FIG.2. Pressure vessel for depressurization tests.

6.2. Pressure relief and blowdown tests

While the emergency core-cooling tests are primarily expected to yield information on the heat transfer and the thermal behaviour of the reactor core, this project is geared to the simulation of hydrodynamic events during rapid depressurization. Next to the determination of blowdown rates for possible loss-of-coolant accidents, measurements of pressure differentials at vessel internals are of the most interest. Studies on the simulation of water-cooled reactors have been carried out since March 1968 by the Battelle Institute in Frankfurt. Theoretical work at the Technical University of Munich, which is carried out simultaneously with these experiments, is directed towards the development of suitable blowdown computer codes.

After experiments with a pressure vessel made of glass for a maximum pressure of 45 bar (in connection with visual observations of the boiling state) and with a small steel pressure vessel for operating pressures up to 70 bar, a pressure vessel for operating pressures up to 140 bar and with the approximate height of a reactor pressure vessel is available for further experiments (Fig.2). This enables experiments with vessel internals to be performed at actual pressure distribution, which in the model is essentially the same as in the original. In the new pressure vessel, the conditions of both boiling- and pressurized-water reactors will be simulated. It is planned to include vessel internals and, eventually, a pressurizer in these experiments.

6.3. Containment tests

While the projects previously described have the objective of ascertaining heat removal during fast loss of coolant, this project is designed to provide the basis for the integrity of a multi-compartmented containment vessel

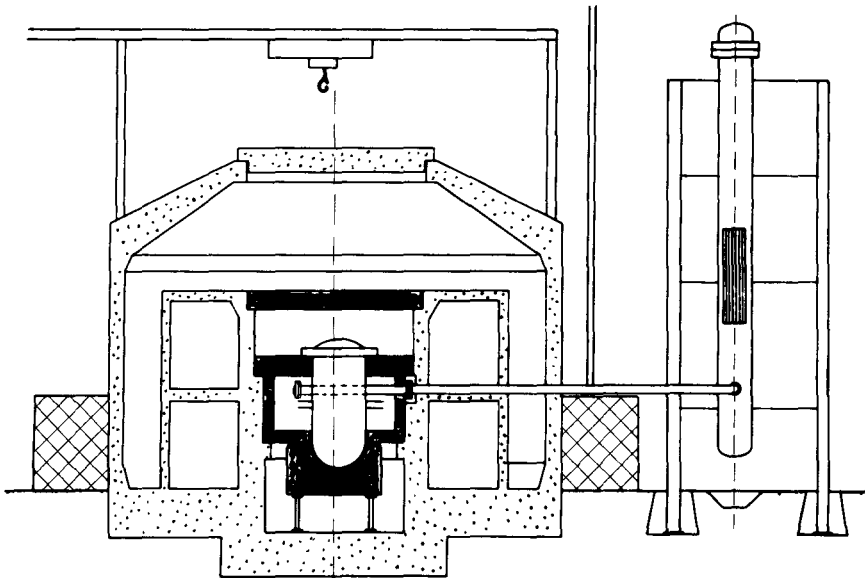


FIG.3. Containment test rig for a PWR.

during a loss-of-coolant accident. For this purpose, the flow conditions in a compartmented full-pressure containment vessel will be investigated by experiments which simulate accident conditions to verify the validity of existing computer codes. These experiments should be a valuable supplement to those conducted so far with pressure suppression systems and in non-compartmented containments. The experiments will be conducted on a one-quarter-scale model containment (Fig. 3), which is similar to the containment of the nuclear power plant Biblis. It is intended to use exchangeable inserts in the immediate surroundings of the pressure vessel to simulate different situations in boiling- and pressurized-water reactors. A pressure vessel already in existence will be used to simulate the reactor pressure vessel.

At this point a few general objectives might be mentioned:

- (1) Investigation of the thermodynamic and flow-mechanical conditions (temperature and pressure histories in the different compartments, flash forces, conditions during condensation, effectiveness of spray cooling systems, etc.).
- (2) Investigation of stresses on structural parts (local pressure distributions, strains, accelerations, behaviour of concrete block walls and of reinforcements, etc.).

The projects as described in sections 6.1 to 6.3 represent the focal point of the investigation of loss-of-coolant accidents with regard to the experimental and financial effort involved. These investigations will be supplemented by a number of further research projects.

6.4. Studies with liquid models

Experimental work with Frigen has been in progress for some time to contribute to a solution of the burn-out problem.

Within the framework of a research project at the Technical University of Hannover, the laws of similarity for heat transfer and hydraulic problems of two-phase flow under rapid transient conditions will be investigated under conditions prevailing during a loss-of-coolant accident or after failure of the recirculation pumps.

To be able to extrapolate the emergency cooling experiments with single bundles to the geometry of a nuclear reactor (a multitude of fuel rod bundles), an attempt will be made to conduct small-scale multi-bundle experiments with a suitable liquid model.

6.5. Computer codes

A few computer codes for the description of all phases of the loss-of-coolant accident are currently being developed and the program is to be extended.

6.6. Failure of emergency cooling

In separate studies conditions during meltdown of an uncooled reactor core and various methods of heat removal from the molten core will be investigated.

In addition to the investigation on the consequences of a loss-of-coolant accident, there are other research projects on various aspects of reactor safety. A few of these are listed below.

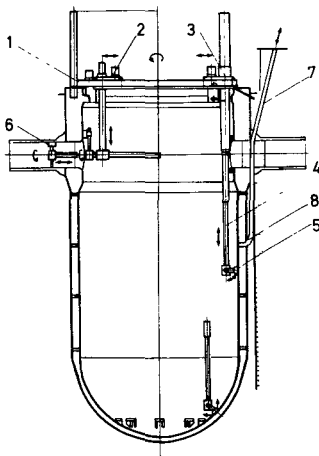
Studies for the improvement of pressure vessel inspection and of fracture mechanics:

investigation of the acoustic waves emitted during crack propagation in pressure vessels, as a means of controlling crack propagation;

research projects in the field of fracture mechanics;

extension of fracture mechanics with the object of applying it to all reactor components, evaluation of the characteristic fracture-mechanical data of materials with tests on a forged ring of a reactor pressure vessel, experimental verification of results of calculations for brittle and ductile fracture with tests on small pressure vessels or pipes;

development of non-destructive testing techniques to discover faults in thick-walled reactor pressure vessels (method of non-destructive recurring tests with an ultrasonic probe system (cf. Fig.4 for accessibility and mobility).



- 1 ROTATABLE PLATFORM WITH SUPPORTS GUIDE PLATE
- 2 SUPPORT FOR NOZZLE TESTING EQUIPMENT
- 3 SUPPORT FOR PRESSURE VESSEL TESTING EQUIPMENT
- 4 TEST ARM FOR PRESSURE VESSEL TESTING EQUIPMENT
- 5 MOVABLE TEST HEAD PRESSURE VESSEL TESTING
- 6 MOVABLE TEST HEAD NOZZLE TESTING
- 7 GUIDEWAY FOR TEST EQUIPMENT
- 8 TEST HEAD FOR FAULT DETECTION

FIG.4. Arrangement for ultrasonic testing of the pressure vessel.

Development of an anthropomorphic synchro-telemanipulator for use after radiation accidents.

Inclusion of extreme accidents of boiling- or pressurized-water reactors in systematic failure analyses.

Study of site-dependent parameters with the objective of quantifying the safety to be gained by proper siting.

Identification and mitigation of the consequences of severe reactor accidents.

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SAFETY OF NUCLEAR POWER STATIONS AND RESPONSIBILITIES OF THE UTILITIES

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Abstract-Résumé-Аннотация-Resumen

SAFETY OF NUCLEAR POWER STATIONS AND RESPONSIBILITIES OF THE UTILITIES.

The special circumstances relating to nuclear power have already resulted in strict control by different authorities, mostly on a national basis. Decisions on technical problems can, however, be made on a more local basis, but the overall responsibility and policy remain with the utility both formally and in practice. The utility carries the legal responsibility towards the authorities that the station will comply with current regulations and towards the employees and the general public if damage of various kinds should occur. Only the utility as the operator of the station has the direct primary information on the state of the station throughout its whole lifetime. With this background, examples are given of the approach of the Swedish State Power Board in specific cases. Among these are mentioned, in particular, the release policy for gaseous radioactivity and the principle of complete separation between redundant trains in safety systems which implies features in the layout and structure that go beyond most current practices.

LA SURETE DES CENTRALES NUCLEAIRES ET LA RESPONSABILITE DU MAITRE D'ŒUVRE.

Les caractéristiques de l'énergie nucléaire ont déjà conduit à un contrôle sévère, exercé par différentes autorités sur une base principalement nationale. Des décisions satisfaisantes concernant les problèmes techniques peuvent cependant être prises sur un plan plus local, mais la responsabilité générale et l'application de la politique d'exploitation reviennent au maître d'œuvre aussi bien du point de vue formel que pratique. Le maître d'œuvre est également responsable envers les autorités de l'observation des règlements en vigueur, et vis-à-vis du personnel et du public en cas de dommages divers. Seuls le maître d'œuvre et l'exploitant possèdent les informations primaires directes sur l'état de la centrale pendant toute la durée de son service. Sur ces bases, quelques exemples sont donnés sur l'attitude observée par la Direction nationale de l'énergie dans des cas spécifiques. Parmi ceux-ci, on peut mentionner en particulier la politique concernant l'évacuation de gaz radioactifs et le principe d'une complète séparation des redondances dans les systèmes de sécurité, ce qui impose à la disposition et à la structure des caractéristiques qui s'écartent de la pratique courante.

БЕЗОПАСНОСТЬ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ И ОТВЕТСТВЕННОСТЬ УПРАВЛЕНИЙ ЭНЕРГОСИСТЕМ.

Особые обстоятельства, связанные с развитием атомной энергетики, привели уже к тому, что эта отрасль находится под контролем различных управлений энергосистем, в основном в общенациональных рамках. Решения же технических проблем можно вполне принимать в рамках местных условий, однако общая ответственность за обеспечение безопасности и выработки основных принципов в этой области лежит формально и практически на управлениях энергосистем. Управление энергосистемы несет юридическую ответственность перед органами власти за то, чтобы станция удовлетворяла всем необходимым требованиям, а также — перед обслуживающим персоналом и окружающим населением по части опасностей, которые могут иметь место при эксплуатации атомной электростанции. Только управление энергосистемы, как владелец атомной электростанции, получает непосредственную первичную информацию о состоянии станции на протяжении всего ее срока службы. Исходя из этого, приведены примеры методов, применяемых Шведским государственным энергетическим управлением в каждом конкретном случае. Среди этих примеров можно упомянуть, в частности, порядок выброса газообразных радиоактивных продуктов и принцип проведения полного разделения многочисленных линий в системах безопасности. Данный принцип воплощает ряд характерных особенностей в самой компоновке и структуре системы, что само по себе выходит за рамки существующей практики.

LA SEGURIDAD DE LAS CENTRALES NUCLEARES Y LA RESPONSABILIDAD DE LA COMPAÑIA ELECTRICA.

Las circunstancias especiales en torno a la energía nuclear han desembocado en un control estricto ejercido por distintas autoridades, que en la mayor parte de los casos son de carácter nacional. Sin

embargo, las decisiones sobre problemas de carácter técnico pueden tomarse en un plano más local, pero tanto en el aspecto formal como en el práctico la responsabilidad general y la dirección corresponden a la compañía eléctrica. La compañía responde legalmente ante las autoridades de que la central cumple la legislación vigente y ante sus empleados y el público general, en el caso de que se originasen daños de tipo diverso. Sólo la compañía, como explotador de la central, tiene información de primera mano acerca de la situación de la central durante toda su vida. Desde estos puntos de vista se presenta la postura del Statens Vattenfallsverk de Suecia en casos específicos. Entre ellos cabe mencionar, en particular, la situación con respecto a las emisiones de productos radiactivos gaseosos y el principio de la segregación total en los canales múltiples de los sistemas de seguridad, lo que significa características de disposición y estructura que se salen de las prácticas normales.

1. INTRODUCTION

Nuclear power, especially that generated by light-water reactors, has now been developed to such a stage that, in the majority of industrially developed countries, it can successfully compete with conventional steam and hydroelectric power. Nuclear power has the additional advantages of being harmless to the environment and safe as regards a secure fuel supply. In contrast, thermal power stations fired with oil or coal often cause problems by giving rise to air pollution. Such power stations discharge large quantities of flue gases containing sulphur dioxide, nitric oxide, carbon monoxide and carbon dioxide, as well as a variety of metals, including radium. A further disadvantage is that the increase in the cost of fossil fuels is difficult to predict and such an increase has a serious effect on the cost of electric power, since the fuel cost represents a significant proportion of the total cost of power.

Nuclear power production can therefore be expected to expand rapidly. It is anticipated that countries outside the Eastern Socialist block will be operating nuclear power stations with a total installed capacity of about 300 000 MW by 1980 [1]. In Sweden, we expect to have installed by 1980 about ten nuclear power units with a rating of between 600 and 1000 MW and a total capacity of about 7000 MW. It is anticipated that one unit will be put into operation each year in the middle of the nineteen-seventies and two units per year towards the end of the decade [2].

Although nuclear power has many advantages from the environmental point of view, it involves special safety problems due to the fact that the reactor contains large quantities of radioactive material. There is thus a hypothetical possibility that, in the event of an accident, a plant could release dangerous quantities of radioactive material to the environment. In this respect, nuclear power stations differ from other sources of power, and the utilities are therefore faced with an entirely new situation of which they have no experience. In addition, the historical background accentuates the problem, since the origin of nuclear power is indelibly associated with the development of the atomic bomb. The situation is complicated further by the fact that nuclear power reactors represent an entirely new technique and that they must be very large if they are to be economically attractive.

In the following, we intend to illustrate some of the problems which the utilities must face with regard to the safety of the design and the operation of nuclear power plants. In particular, the distribution of responsibility between the various parties concerned is of vital significance in the realization and operation of nuclear power stations and this matter is therefore given special emphasis. The policy of the Swedish State

Power Board on safety and the influence of safety aspects on the handling of specific problems are also discussed. The problems of radioactive discharge during normal operation are treated and examples are given of the measures taken in the design of certain systems to reduce the potential risks of accidents involving radioactive releases.

2. REGULATING BODIES AND AUTHORITIES

The special conditions associated with nuclear power, especially with radiological safety matters and the handling of fissile material, have given rise to the establishment of regulating bodies and authorities with supervisory duties. The original close association with military applications has probably also been a contributory factor in many countries. However, even in countries where work is done solely on the peaceful applications of nuclear power, the activities are still supervised by central authorities.

Thus, in the majority of countries, authorities have been established for the purpose of ensuring that nuclear power plants are designed in such a manner that the risk of serious accidents will be acceptably small. In addition, the authorities ensure that the plants are designed in such a manner that dangerous radioactive discharge to the surroundings is prevented if an accident should occur. They also control the normal radioactive discharge from nuclear power stations.

Apart from complying with the special regulations of such supervisory authorities, nuclear power stations, like other thermal power stations, must also be designed in accordance with regulations on the safety of pressure vessels, industrial safety, electrical installations, etc. However, owing to the regulating bodies and authorities mentioned above, a utility planning to build a nuclear power station will find the situation extremely complicated. Conditions are rendered even more difficult by the fact that the limits of responsibility of the various authorities are sometimes vague.

The guidelines and regulations considered necessary for the realization of the nuclear power stations in operation today are defined, in practice, somewhat differently in different parts of the world. In spite of significant efforts, particularly in the United States of America, in the fields mentioned above, it must be considered that the position is still far from clear as regards criteria, standards, authorities, attitudes, etc. However, such guidelines are required to the maximum extent possible for the guidance of practical work. On the other hand, the authorities have been careful when establishing definite regulations, with the understandable aim of avoiding unnecessary technical restriction at a stage where the situation has not yet crystallized.

This implies both advantages and disadvantages for a utility that is going to build a nuclear power station. It is an advantage that development is not restricted by excessively stringent regulations, but it is a disadvantage that the technical level which the authorities require of a design before granting approval is often unclear.

Among the activities necessary to complete a nuclear power plant, only a few are formally regulated and are actually suitable for such regulation. The majority of guidelines that should be applied must thus be

determined by the responsible builder. This allows scope for wide variations and has also led to differences in practice in various parts of the world. The appropriate authority must nevertheless always approve the whole station. The approved design can later be employed as reference and thus gains increased importance for future stations.

3. DISTRIBUTION OF RESPONSIBILITY

Since the design of nuclear power stations from the point of view of safety is controlled by special authorities which, in some countries, have issued regulations, doubt may arise as to who bears the final responsibility for the plant and its safety. It is sometimes considered that a utility need only follow the regulations of the authorities in order to have fulfilled its obligations as regards the safety of the station, and consequently, its obligations towards the general public. If, nevertheless, an accident should result in a radioactive release during the lifetime of the reactor, the safety authorities would then be considered responsible for not having issued satisfactory regulations. Such views would be understandable if the authorities possessed the widest knowledge in the field. Utilities which had not been engaged earlier in the field of nuclear energy have therefore employed this principle in their work in some cases.

It is hardly possible to reconcile this approach with prevailing conditions. The utility, being the owner of the plant, must make all final decisions, not least as regards the economic factors. It also serves as the centre for all pertinent information concerning the plant. As mentioned earlier, neither the construction nor the operation of a nuclear power plant is regulated in detail. The directives or guidelines of the utility/owner are therefore necessary in a very large proportion of the overall activities. Practically all activities in conjunction with a power station are related to the direct actions of the utility. The authorities have no direct responsibility for the plant, and should be regarded as providing the guarantee that certain minimum requirements, as yet not definitely specified, are satisfied as concerns the method of operation, systems, components, etc.

It would appear particularly unreasonable for an authority to have any direct responsibility for the safety of the operation of a plant, since only the operating staff of the utility have an overall view of the plant. It would be just as unreasonable for an authority issuing driving licences to be held responsible for the errors of the individual driver as it would be for a safety authority to take direct responsibility for the operation of an individual station. In practice, the central function of the utility undeniably comprises all safety aspects. In many countries, e. g. in Sweden and the United States of America, this is also legally the case.

However, the conditions outlined above do not imply any under-estimation of the safety authorities and their functions. An important function of these authorities is to determine the boundaries of the activities related to the introduction of nuclear power. In principle, this function does not differ from the functions of authorities in other fields, e. g. industrial safety, etc. The responsibility of the other authorities can likewise be considered analogous.

The utility can sub-contract various services, but the governing decisions remain in the hands of the utility itself. Since the utility acts as the responsible party in relation to other interested parties — employees, authorities, customers, general public — it must also be responsible for the commitments of suppliers. This will be especially clear when considering the operation of the plant, with which the suppliers are no longer concerned, and will also define the relationship between the utility and the suppliers. However, it can sometimes be difficult for suppliers to understand this relationship completely. Experience shows that in practice, these problems cannot be ignored and their consequences will become more severe the longer the solutions are delayed.

The reference and evaluation backgrounds of the supplier and the purchaser are, and will remain, different because their responsibilities are different.

It can be seen that a pure 'turnkey' plant where the governing decisions are made by the supplier would appear to be incompatible with the pattern of responsibility outlined above. A nuclear power station on a 'turnkey' basis must therefore be regarded as an exception, possibly suitable for a utility which had not been engaged earlier in the field of nuclear power and which must therefore take some risks while acquiring experience. A purchaser without at least the same level of competence as the supplier is more or less at the mercy of the supplier. If the 'turnkey' purchase is nonetheless carried through, the consequence would appear to be that the utility basically only acts as the financing body and that the actual responsibility for the plant will rest with another party, e.g. the supplier. This should be regarded as an undesirable condition for both the general public and the utility.

4. POLICY OF THE STATE POWER BOARD AND SOME APPLICATIONS

The responsibility of the utility for safety and the motives for retaining the utility as the policy-maker have been outlined above. In Sweden, we have attempted to follow largely the regulations laid down in the United States, primarily by the USAEC, since great progress has already been made in the United States in this field. The Swedish State Power Board also collaborates continuously with the Swedish authorities so that the policy applied within the State Power Board is based on practical experience gained both at home and abroad, while at the same time the intentions of the Swedish authorities are taken into account.

The basic aim is to obtain the best technical and economic solutions to problems concerning nuclear power stations, maintaining the necessary level of safety. In this context, it is important to point out the necessity of defining, at the earliest possible stage, the safety requirements or safety policy to be applied to a nuclear power station to avoid unnecessary, and often very expensive, modifications at a later stage. If the requirements are specified from the beginning, a technical solution can often be found within the existing time schedule and without significant extra cost. The safety analysis of a nuclear project should be co-ordinated with all other activities of the project, so that the functional requirements of systems and components are known prior to design and procurement.

It may now be appropriate to present some examples.

Bearing in mind the anticipated large expansion of nuclear power, many have found it advisable to tighten the rules concerning radioactive discharge during normal operation, which are based on the radiation doses recommended by the ICRP. Thus, the Swedish radiation protection authorities have recommended that in A. D. 2000, the normal radioactive discharge from nuclear power plants, including plants for the processing of fuel, shall not be responsible for more than about 6% of the permissible dose (recommended by the ICRP for large population centres) and that the remainder shall be reserved for other activities and shall provide a margin for discharge in the event of accidents [3]. This implies that, in A. D. 2000, the Swedish population will be exposed to a radiation dose not exceeding an average of 10 mrem/yr from the nuclear power industry.

Since Sweden will then have about 100 nuclear power units and the normal gaseous discharge from these units may account for about one-half of the permissible radiation dose, the permissible gaseous discharge from each unit will be equivalent to an average radiation dose of about 0.05 mrem/yr, taken over the whole of the population. An analysis of radioactive discharge from BWR and PWR plants shows that the latter can readily meet these requirements with a comfortable margin of safety, whereas in the case of BWR reactors, hold-up of radioactive discharge in excess of the amount hitherto normal is necessary.

A further rule is that each radiation dose is to be regarded as being undesirable and shall be avoided if it is economically feasible. According to the calculations carried out by the National Institute of Radiation Protection, each measure aimed at reducing the radiation dose is justified. It is suggested that such measures should be implemented if the cost amounts to no more than about \$ 200 per man-rem reduction [3].

It is interesting to note that these recommendations have the same practical consequence as the rules issued by the AEC in 1970, i. e. that the radiation exposures to the public from a nuclear plant shall be "as low as practicable" and shall not exceed small fractions of the exposure recommended in the ICRP guides [4].

As a result, the State Power Board aims to select a design radiation dose from each large nuclear power unit, at the boundary of the safety zone, of about 1% of the value permissible according to the ICRP. By this means, the margin of safety will be of such magnitude that it shall be possible to maintain discharge far below the values specified by the ICRP, even if the discharge from one of the stations should temporarily rise above the design value, due to some fault in the plant.

These principles will imply that the State Power Board will provide its future BWR plants with filters which shall be capable of retaining the noble gases for a sufficient length of time to allow them to decay to a low value.

In view of the desirability of ensuring that the probability of accidents involving the risk of radioactive discharge shall be as low as possible, the State Power Board has drawn up guidelines for the design of a plant. Some of these are not commonly used in other plants and are discussed here.

The State Power Board applies the principle that redundant chains of components in emergency systems shall always be physically separated in the manner described below. The physical separation shall be so thorough that mishaps such as fire or floods shall not be capable of

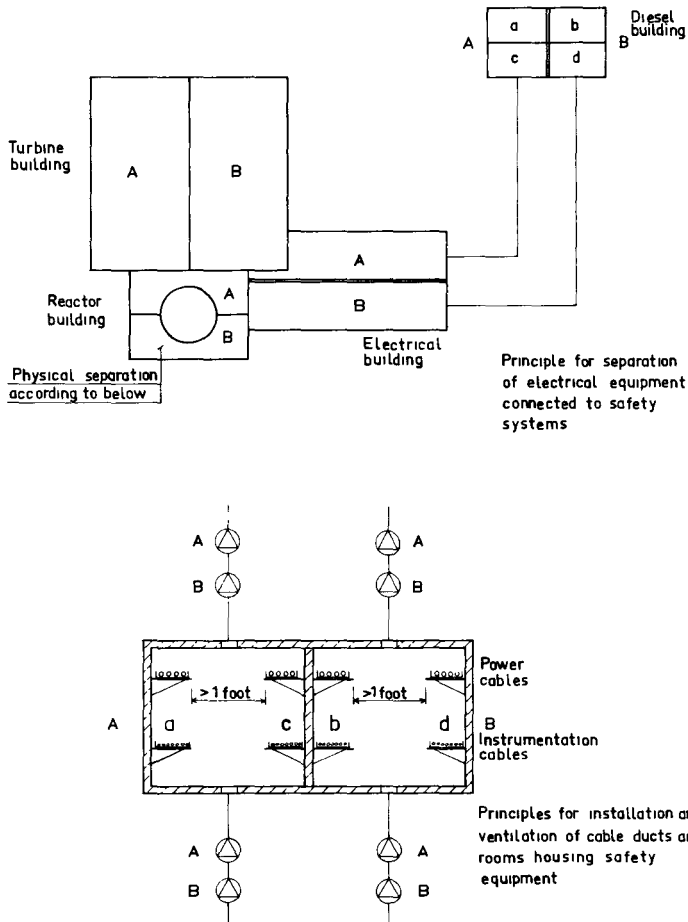


FIG. 1. Principal arrangements of electrical cable separation and installation.

rendering inoperative more than one chain. This implies that the various redundant chains are installed in separate parts of the building, so that one of the chains can be flooded or damaged by fire without affecting the other chain. In view of the fact that water which can damage a variety of equipment is present all over a station, this principle calls for careful design of the station and the layout of the systems.

To ensure that the fire protection system is reliable, the electric power supply to the various redundant chains must be separated (see Fig. 1). For example, in the Ringhals station this has been solved by dividing the power supplies into two cable runs, basically with four busbars, i. e. two busbars per cable run. The emergency power supply is provided by four diesel generator sets, connected directly to the four busbars. Each cable run is taken through a separate cable culvert and the two busbars included in each cable run are as far apart as possible in the culvert. The principle of division into separate culverts has been applied to each

floor in the station building and each of the cable runs terminates at a separate electrical room. However, for structural reasons, the distribution on each floor cannot be separated equally strictly. In cases where it is not practicable to separate the cables, they are run in steel conduits to prevent the spreading of a possible cable fire. It should also be mentioned that the two separate cable runs, both in culverts and in electrical rooms, will be separately ventilated to ensure that fire cannot spread through the systems, that the temperature in these spaces will not rise to an excessively high value and that the highly corrosive gases formed in the event of a fire do not affect any other electrical equipment.

A further principle which is employed with few exceptions is the 'half-hour' rule. This implies that all measures which must be taken within half an hour in the event of abnormal operating conditions, to ensure that the safety of the station is not jeopardized, shall be automatic. However, this does not mean that the principle precludes manual remedial measures, but rather that it is considered to be insufficiently reliable from safety considerations to rely on the operational personnel being capable of making quick, accurate decisions and carrying out the correct remedial measures in critical situations within a relatively short period of time.

A further factor which has a significant influence on the safety of a nuclear power station is the accessibility of systems and components. Efforts to ensure good accessibility for inspection, testing and repair of the equipment are made by devoting appreciable effort to obtaining good structural and system layouts, with extensive separation of components in separate rooms. In addition, extremely low leakage from components and thus a low level of radioactivity are required, and this is achieved by measures such as the extensive use of welded joints in the primary system or the employment of double seals with intermediate drains connected to a drainage system operating under vacuum.

The ventilation system is also an important factor with regard to access to equipment located in different rooms. Even in this case, appreciable efforts are made to provide separate ventilation for the various areas and rooms, so that radioactive leakage in one area is not 'ventilated' into another.

A further principle applied by the State Power Board is that of flooding with sea water as a last resort in the emergency cooling of some types of reactor. In the event that the emergency cooling system should become inoperative after an accident, provision is made for manual admission of sea water as a direct coolant and even pumping sea water into the containment to flood appropriate parts of the plant. It should be noted in this context that the contact between sea and station is open in one direction only, so that radioactive water is prevented from escaping through the system. On some types of reactor, such as PWR, this principle cannot be employed, since the reactor normally operates on borated water, and if the core were filled with ordinary sea water, a reactivity accident could occur.

In the event of a major pipe rupture outside the containment, the building where such a break takes place will have to accommodate large steam and water flows. It is easily shown that, without specific provisions, an ordinary building would suffer serious damage with the definite risk of damaging or destroying, among other things, safety equipment located

in that building. The provision of the State Power Board consists of special relief areas in suitable walls, etc., so that the large steam flow can easily be discharged without causing too large an internal pressure to build up in the different rooms of the building.

Finally, the significance of sound general technical provisions and quality requirements should be mentioned. In these matters, it is especially important that the purchaser and supplier are aware of each other's quality requirements before the contract is signed. If there is incompatibility, as there always appears to be, agreement should be reached by discussions, possibly backed up by price adjustments. These are normally larger if the changes are made after the contract has been signed. It should be pointed out that general specifications and quality requirements should not be such as to necessitate the use of newly designed, unproven components.

5. THE UTILITY AND THE GENERAL PUBLIC

A nuclear power station will always affect its surroundings, since, even under normal operation, it will discharge radioactive material. In addition, there is the potential risk, small though it is, that the station will discharge activities as a result of an accident in such quantities that the population in the surroundings will be affected. The population of the region is therefore justified in demanding reliable information concerning the consequences of building a nuclear energy plant.

Bearing in mind that the safety authorities in most countries have hitherto placed restrictions on the population density around nuclear power stations, it is natural that the general public draws the conclusions that some risks must be associated with nuclear power, since restrictions would otherwise not be required. It is therefore essential for utilities to provide the general public with satisfactory information on these conditions.

Before construction of a plant is started, the motives behind the plant and its value should be explained to the population in the vicinity of the projected station. The significance of the increased radiation dose in relation to the natural background radiation must be clarified. It must then be emphasized that the plants are being designed for very small radiation exposures as compared with the background radiation and the limiting doses recommended by the ICRP.

A comprehensive and clear presentation of the great advantages of nuclear power from the environmental point of view as compared with other sources of power is essential. The risks of radioactive discharge in the event of an accident should then be thoroughly discussed, showing how small these risks are and placing them in relation to other risks in the community. It should be clarified that the overall risk to the public can be reduced by the rational use of nuclear power to provide the large quantities of energy required for further development in a healthy and risk-free manner which is harmless to the environment. Open and clear information is the best means for utilities to gain the confidence of the general public.

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SAFETY OF NUCLEAR POWER PLANTS

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Abstract—Résumé—Аннотация—Resumen

SAFETY OF NUCLEAR POWER PLANTS.

The operation of the Central Electricity Generating Board's nuclear power stations is such that defined safety standards are adhered to. This requirement is related to operation under normal (non-fault) conditions, and to the unlikely condition of a significant plant fault which might lead to fission product release into the pressure circuit and subsequently to the atmosphere. The design of the stations must ensure that operation at the design power permits the safety standards to be maintained. Design standards are outlined with particular reference to the steel pressure vessels and the development of the concrete vessels. The protection standards of independent parameter tripping with adequate redundancy are described and the development of later system designs given. The extent of surveillance during construction is discussed to ensure the design standards are achieved. The plants are operated within the criterion that, in the event of the maximum credible accident occurring, the probability of a fuel can temperature in the reactor greater than 640°C does not exceed 0.01. Operating conditions consistent with this standard are defined in Operating Rules. Plant Maintenance is summarized in a Maintenance Schedule and the significance of this maintenance in ensuring satisfactory plant conditions is described. Finally, examples of unusual incidents which have occurred are given.

SECURITE DES CENTRALES NUCLEAIRES.

Les centrales nucléaires du Central Electricity Generating Board sont exploitées de telle sorte que des règles de sécurité bien définies soient observées tant dans les conditions normales (pas de défaut) que dans le cas improbable d'un défaut qui pourrait entraîner un dégagement de produits de fission dans le circuit sous pression et, de là, dans l'atmosphère. Les centrales doivent être conçues de manière que l'exploitation à la puissance prévue permette de respecter les normes de sécurité. Le rapport indique les normes de construction, notamment pour les cuves de pression en acier et le développement de caissons en béton. Il expose la philosophie de la protection qui fait appel à des paramètres indépendants (et en nombre adéquat) pour le déclenchement des sécurités et rend compte de la mise au point de systèmes plus récents. Les auteurs examinent le degré de surveillance nécessaire pendant la construction pour assurer que les prescriptions soient satisfaites. Le critère pour l'exploitation du matériel est le suivant: au cas où l'accident le plus vraisemblable se produit, la probabilité que la température de la gaine d'un élément combustible dans le réacteur dépassera 640°C n'est pas supérieure à 0,01. Les consignes d'exploitation définissent des conditions d'exploitation compatibles avec cette norme. L'entretien du matériel fait l'objet d'un Guide d'entretien, et le rapport montre l'importance que revêt cet entretien en assurant des conditions d'exploitation satisfaisantes. Les auteurs décrivent, enfin, des exemples d'incidents insolites qui se sont produits.

БЕЗОПАСНОСТЬ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ.

Эксплуатация атомных электростанций Центрального энергетического управления СК включает выполнение определенных требований норм безопасности. Эти требования относятся к эксплуатации станции как в случае нормальных (безаварийных) условий работы, так и в случае возникновения серьезных аварий, которые могут привести к выбросу продуктов деления в контур, находящийся под давлением, а затем — в атмосферу. Конструкция атомной электростанции должна обеспечивать ее эксплуатацию на проектной мощности с неременным соблюдением норм безопасности. Проектные нормы особенно четко определяются в отношении к стальным корпусам давления и бетонным корпусам. Приводятся стандартные требования, предъявляемые к независимым параметрам выключения реактора с учетом соответствующих превышений. Описаны последние разработки конструкций систем. В докладе обсуждается степень надзора, необходимого в процессе сооружения станции для уверенности в выполнении проектных стандартов. Определен критерий эксплуатации станции, который устанавливает, что в случае возникновения максимально возможной аварии вероятность роста температуры топливной оболочки выше 640°C не должна превышать 0,01. Эксплуатационные условия с учетом этой нормы определяются правилами эксплуатации. По-

рядок обслуживания станций кратко изложен в графике планово-предупредительного ремонта, где указана также важность такого обслуживания для обеспечения удовлетворительных условий работы станции. Наконец, приводятся примеры необычных случаев аварий, которые имели место.

SEGURIDAD DE CENTRALES NUCLEARES.

La explotación de las centrales nucleares del Central Electricity Generating Board se lleva a cabo de modo que se respeten ciertas normas de seguridad. Este requisito se refiere tanto al funcionamiento en condiciones normales (sin averías) como a la situación improbable de una avería significativa de la central que pueda conducir a la liberación de productos de fisión en el circuito de presión y posteriormente en la atmósfera. El diseño de las centrales debe asegurar que su funcionamiento a la potencia nominal permite que se cumplan las normas de seguridad. Se esbozan las normas de diseño, refiriéndose en particular a las vasijas de presión de acero y al desarrollo de las vasijas de hormigón. Se describen las normas de protección basadas en el disparo provocado por parámetros independientes con la redundancia adecuada y se expone el desarrollo de los últimos sistemas de protección. En la memoria se discute el grado de vigilancia necesario durante la construcción para asegurar que se cumplan las normas de diseño. El criterio para el funcionamiento de la central es que en caso de ocurrir el máximo accidente verosímil, la probabilidad de que en el reactor la temperatura de vaina sobrepase 640°C no debe ser mayor que 0,01. Las condiciones de funcionamiento consistentes con esta norma están definidas en las Reglas de Operación. El entretenimiento de la central está resumido en un programa específico y se describe la importancia de este entretenimiento para asegurar unas condiciones satisfactorias de la central. Finalmente se exponen ejemplos de incidentes no usuales que han ocurrido.

1. INTRODUCTION

The Central Electricity Generating Board in the United Kingdom has the responsibility for the supply of electricity in England and Wales. This includes acceptance of the design and construction of power stations, their operation and high voltage transmission. It is also responsible for forward planning in electricity supply, in conjunction with the Government.

Table I summarises the system at present operating.

The nuclear component consists of eight twin Mark I Gas Cooled Reactor (Magnox) stations, each utilising natural uranium fuel canned in Magnox, with a graphite moderator and cooled by pressurised CO_2 . Main features of these stations are given in Table II.

For Berkeley – Sizewell, the pressurised CO_2 is contained in a steel pressure vessel ($\sim 20\text{m}$ diameter), connected by large (2m diameter) steel ducts to external boilers; the Oldbury and Wylfa systems are integrated reactor/boiler units contained in pre-stressed concrete pressure vessels of internal diameter of approximately 30m.

The future proposed programme increases the relative proportion of nuclear capacity, and by 1980, it is expected that the total system will be about 80 GW of which 17% will be nuclear.

The increase in nuclear capacity arises mainly from the Mk. II Gas Cooled Reactor (AGR) concept, utilising enriched UO_2 fuel clad in stainless steel, with a graphite moderator and cooled by CO_2 in a pre-stressed concrete vessel.

It is evident that the CEGB is responsible for a large amount of nuclear generation, and the safety of this plant is extremely important. The regulatory body (Nuclear Installations Inspectorate, responsible to the Secretary of State for Trade and Industry) requires to be satisfied on safety aspects before they give their approval for operation. Independently of this requirement the CEGB sets very high safety standards, and has the additional motive of protecting their large capital investment.

This paper reviews how safety standards have developed, and have been applied to CEGB nuclear stations. The main concern of the paper will be with the Mk. I GCR stations, but some reference will be made to design developments for the later stations. Operational procedures for the Mk. I GCR stations ensuring operation within the safety criteria are discussed, and examples of unexpected occurrences with safety implications are given.

TABLE I
CEGB SUPPLY SYSTEM (1970)

Type of Fuel	Installed Capacity	% of Installed Capacity
Coal	38 GW	78
Oil	6 GW	12
Nuclear	3 GW	6
Other	2 GW	4

TABLE II
CEGB NUCLEAR STATIONS

Station & Year Commissioned	CO ₂ pressure kN/m ²	Thermal power per reactor MW	Electrical output per reactor MW
Berkeley 1962	962	556	138
Bradwell 1962	1014	445	125
Hinkley Point 'A' 1965	1379	890	230
Trawsfynydd 1965	1758	670	185
Dungeness 1965	1951	625	205
Sizewell 1966	1930	700	210
Oldbury 1967/8	2517	595	200
Wylfa 1971	2758	1370	431

2. SAFETY STANDARDS

Table II shows that CEGB nuclear stations have been in operation since 1962, which implies construction taking place over more than a decade. Developing technology has led to higher fuel ratings, increased efficiency and higher powers and electrical output as subsequent stations have been introduced. Safety standards have been maintained, and in some areas improved with reactor development. The developments in technology, and the maintenance and improvements in safety standards are related to the requirements of cheaper electricity; thus the provisions made to maintain or increase plant safety have to be balanced against the cost of providing the related equipment. Greater safety can be provided at increased cost and the designer and safety assessor have the difficult task of deciding when 'sufficient' features have been introduced into the design to have achieved a suitable level of safety.

TABLE III
MAIN FAULT CONDITIONS – MK. I G.C.R.

Fault Condition	Arising From:	Comment
Reactivity increase (a) Symmetrical (b) Asymmetrical	Bulk rod movement Sector rod movement	] These are considered as credible faults, both individually and as a bulk group + sector group rod movement.
Loss of Coolant (a) Loss of Flow (b) Loss of Pressure	Single or multiple blower failure Pressure Vessel Failure Pressure Circuit (duct or boiler) Failure	
Loss of Feed Water Flow	Pump failure or pipe fracture	Both considered as credible fault conditions.
Channel blockage	Debris in channel	Credible fault condition.

The criteria for assessing the Mk. I GCR plants are:

- (i) During normal operation the radiation dose received by operating personnel shall be minimised as far as practicable, and shall not exceed 5 rem per year (with a maximum of 3 rem in any three calendar months),
- (ii) During normal operation, the dose received by members of the general public living in the vicinity of the plant shall be minimised as far as practicable and shall not exceed 0.5 rem per year,
- (iii) Under accident conditions, the fission product release to atmosphere shall be such that members of the general public shall not receive doses in excess of limits recommended by the Medical Research Council with a predefined probability. For CEGB plants, the dominant effect in most cases would be due to ^{131}I thyroid dose; administration of stable iodine and evacuation of members of the public may be called upon to meet this criterion,
- (iv) Routine discharges of radioactivity arising from plant operation, e.g. gaseous and liquid effluent shall be minimised (and are subject to Government authorisation).
- (v) Monitoring of the environment of the plant shall be undertaken.

Items (i), (ii), (iv) and (v) are discussed in [1]. The standards of design and operation in relation to item (iii) will be developed in this paper.

The method of safety assessment used for the Mk. I GCR plants is defined. The approach adopted has been the credible/incredible accident analysis in respect of the initiating fault conditions; reliability analysis has been used in some cases for the performance of safety equipment, and essential auxiliaries. The reasons for adopting the credible/incredible approach were historical since when the original assessments were carried out, this was the only approach available. Probability and credible/incredible have their merits and demerits and ultimately both rest on the judgment of the designers and safety assessors.

3. SAFETY ASSESSMENT AND DESIGN PRINCIPLES

3.1 Potential faults

The design features of the Mk. I GCR's are well-known, and it can be seen that Table III summarises the main faults which are assessed. If the reactor is not tripped quickly enough when a fault condition occurs, fuel element temperatures would rise, ultimately causing Magnox melting and release of fission products into the reactor circuit. For depressurisation faults this would almost certainly lead to release of fission products into the atmosphere; for contained faults, leakage into the atmosphere is likely to occur on a relatively slow timescale unless massive uranium melting led to pressure circuit failure.

The table indicates that of the fault conditions listed, 'severe pressure vessel failure' is considered to be incredible, and the remaining faults 'credible'. For these credible faults adequate tripping reliability is necessary, and appropriate auxiliary circuits, for providing cooling after trip, required. Specific items will be reviewed from Table III, to indicate (i) how design requirements have evolved in respect of adequate tripping reliability, and (ii) the maintenance of construction standards in order that vessel failure incredibility can be accepted.

3.2 Reactor tripping

In order to shut a reactor down quickly enough, it is essential that automatic tripping be provided and that reliability of (i) the tripping signals and (ii) rod entry to take place, must be extremely high. The standards for trip initiation adopted in the Mk. I GCR's were:

- (i) No two failures of the protective equipment shall inhibit correct operation,
- (ii) Wherever reasonably possible, at least two separate and physically different parameters shall be used to provide initiating trip signals,
- (iii) Spurious tripping shall be reduced to an acceptable frequency.

These principles have led in general to the provision of two lines of protection, each consisting of three sensors connected in a two-out-of-three matrix, and operating into guard lines in a two-out-of-three matrix.

Although some complicating factors arose because of the requirement to provide automatic protection over a wide range of system states (e.g. power level), and over a wide range of fault conditions implied in any one defined fault (e.g. very slow to very fast depressurisation), it proved relatively easy to meet these standards in most cases. Some specific points follow.

3.2.1 Rapid depressurisation

The fault considered is failure of the pressure envelope outside the main vessel. Analysis of gas flows and reactor transients showed that the worst condition arose from a complete severance of a bottom duct at a specific position which led to stagnation of flow in the core and a rapid rise in fuel can temperature, approximately 10°C per second. For the earliest designs, it had been intended to obtain protection against this fault by can temperature tripping and rate of change of gas pressure signals operating into the guard lines and releasing the normal control rods. It was shown that the time of insertion of the rods (15 seconds) coupled with the delay in tripping off the can temperature signals, resulted in too large a transient. Consequently, several of the 'normal' control rods were replaced by 'Secondary Shutdown (SSD)' rods which included individual rapid acting pressure sensitive release devices. These provided a significantly faster shutdown system against this fault. Lines of protection originally provided for this fault were:

- (i) 2/3 SSD rods + bulk rods tripped from rate of change of pressure.
- (ii) 2/3 SSD rods + bulk rods tripped from can temperature.

The analysis of the fault condition is pessimistic in assuming the worst hole size in the worst position in the duct, coupled with continuing full speed operation of the main blowers.

Although the immediate short-term transient pre-supposes a very unlikely set of circumstances, duct failure requires consideration also of the 'long-term' transient. Duct failure could lead to air ingress with subsequent exothermic air-graphite reaction. If no special measures were adopted a potentially hazardous situation could arise, and provisions are thus made for maintaining emergency cooling including injection of CO₂.

The original assessment considered that failure of rods to enter the core, having received a trip signal, was incredible – although for the burst duct condition it was pessimistically assumed that only 2/3 of the SSD's release mechanisms were effective. Supporting arguments relied on assessment of geometrical configurations and predictions of core stability, plus routine monitoring of rod performance. For Mk. II GCR's currently being constructed, diverse shutdown systems are being provided. The situation differs from that on the Mk. I because of an additional gas baffle through which the top entry rods have to pass and hence there is some slight chance that there may be interference. Investigations are also being undertaken to see whether a diverse shutdown system can be introduced into the Mk. I GCR's specifically for certain depressurisation faults. These may be fitted if, in the future, steel oxidation effects cast any doubts upon the original reliability of top entry shutdown rods. The system being investigated utilises small boron steel balls held in hoppers above selected channels in the reactor core, which can be released when necessary.

Provision of a manually operated boron powder injection system for maintaining shutdown in Mk. I GCR's is being investigated. It could be used either when insufficient normal control rods had entered the core under accident conditions to maintain the system shut down indefinitely, or in the even more unlikely event that rod entry under normal operational conditions became impossible.

3.2.2 Sector rod fault protection

In this area it has not proved possible to provide independent parameter tripping in all cases.

In Mk. I GCR's, instabilities can arise, and operations, such as on-load re-fuelling can induce non-stable conditions – hence the requirement for sector rods, which also assist in overall con-

trol. These rods operate off suitably averaged channel gas outlet thermocouples to maintain temperatures in (normally) nine sectors constant. Faults in the control circuitry could cause a sector rod to motor out, and with the remaining sector rods maintaining control correctly, a localised fault could develop. This produces the paradox that malfunctioning of the system can produce the adverse effect which in the absence of malfunction the system is designed to mitigate.

Protection against this fault is complicated since:

- (a) The only easily available sensor is temperature, from fuel can or channel gas outlet thermocouples,
- (b) The only other fully satisfactory sensor would be in-core flux measuring equipment. Even if suitable detectors of this type were available, use of this equipment to the extent required would be very expensive,
- (c) The fault is particularly difficult to define – it could range from a slow local reactivity increase to rapid rod withdrawal. The position of the sensors and trip margins required depend on the conditions assumed in respect of rate and starting condition (e.g. position of the faulty rod and the remaining sector rods),
- (d) The theoretical analysis of this fault condition is difficult, ideally requiring a three-dimensional time dependent analysis.

Protection has been demonstrated using two lines of two-out-of-three temperature sensors, so that although the principle of diversity of sensors has not been maintained, extra redundancy is incorporated to compensate.

It should be noted however that for some of the Mk. I GCR's, the operators have chosen to maintain the sector systems under manual control. The response is slower than under automatic operation and this eases the protection requirements. The consequential loss of flexibility on reactor control has been small.

For the Mk. II GCR's, sector control will be provided. Potentially the protection problem is considerably simpler than for the Mk. I GCR's, since the safety margins are larger and two channel gas outlet thermocouples per channel are provided. A simple 'auctioneering' system from signals in all channels in a sector may be provided to initiate a reactor trip.

3.2.3 Channel blockage protection

Although the possibility of channel blockage (leading probably to single channel melt-out) is very remote, the Mk. I GCR's are currently being fitted with equipment designed to provide a reactor trip should blockage occur. Experimental evidence suggests that there is a margin in time/temperature between melting and ignition of Magnox in CO₂ and some support for this result has been obtained at St. Laurent and Chapelcross. Although the safety of the reactors will be increased with this modification, it should be stressed that the devices are being fitted for economic and not safety reasons [2]. We have accepted that a single channel melt-out is very remote, but in any case, would not lead to a public hazard for the contained reactor, and thus no additional safeguard is necessary.

The sensors are delayed neutron detectors positioned around the gas outlet ducts and operating into the normal guard lines. It has been demonstrated both theoretically and experimentally, that should channel blockage occur, only a small area of a fuel element in the channel will be exposed through Magnox melting before the reactor is tripped by this equipment. No uranium melting will occur, and hence although the gas circuit will be subject to a significant fission product release, the system will be recoverable as compared with a massive meltdown which might otherwise occur.

4. DESIGN AND CONSTRUCTION STANDARDS – PRESSURE CONTAINING PARTS

The design and construction of the pressure containing parts, for the early Mk. I GCR's with steel vessels was to the appropriate British Standard Specification (BS 1500). For later

stations the Standard written specifically for nuclear reactor steel pressure containing parts BS 3915 has been used. The latter code has additional requirements on material inspection. Further details on these British Standard requirements can be found in [3] and [4].

As no code existed for pre-stressed concrete pressure vessels appropriate notice was taken in the design of code requirements for concrete structures and pre-stressing, and with the steel pressure containing parts being designed to one or other of the steel vessel standards. A British Standard Specification for pre-stressed concrete pressure vessels for nuclear reactors is now in draft form.

The CEBG employed independent agents to assess the design and to inspect all stages of the fabrication of the steel vessels. This arrangement was continued for the steel pressure containing parts of the earlier concrete vessels and in addition an independent consultant was employed for the pre-stressed concrete plant. More recently the inspection duties of the agents have been fulfilled by specialist services within the CEBG under a policy of building up internal know-how and self sufficiency.

In all cases pressure circuits have undergone an overpressure test prior to operation: for the steel vessels this was 1.5 times and for concrete vessels 1.15 times the design pressure.

The introduction of the prestressed concrete vessel led to a revision in the siting policy of the Government; it became accepted that reactors would be sited significantly nearer urban areas than had been the case formerly [5].

The redundancy of prestressing tendons and continual inspection gives great confidence that adequate warning of possible failure will be apparent and a critical reappraisal of standards of design for closures to the vessel was carried out to maintain confidence over the whole pressure system. Depending on the design, closures may be required to be of large dimensions e.g. man access, refuelling standpipes and podded boiler removal where appropriate.

Although much discussion has gone on regarding the design of such closures, the principle of a double closure is generally being maintained. Additionally:

- (a) Each closure shall be independently secured to the main vessel,
- (b) Failure of either closure shall not lead to significant structural damage to the other,
- (c) Methods of monitoring the condition of either closure shall be available during normal operation.

These developments are tending towards the (safety) philosophy for pressure containing structures that the integrity of the structure is not in doubt following failure of any single item.

The experience so far with the concrete vessels at Oldbury and Wylfa has been satisfactory. Some problems were experienced initially with regard to the performance of the insulation in specific areas, but these were overcome. The predictions and observations of vessel movement and long-term creep effects have been in close agreement.

5. SAFETY IN OPERATION

Only major aspects related to potential fault conditions on the plant are discussed. Personnel control, health physics, and environmental monitoring are reported elsewhere [1].

The assurance of continuing safe operation of the plants is obtained by ensuring that:

- (a) The reactors are operated with operating conditions and procedures consistent with the requirements of the safety analysis;
- (b) The plant, and auxiliary plant, performance remains consistent with the design criteria; and
- (c) The operators remain familiar with all relevant aspects of operating and emergency procedures.

5.1 Operating rules

In order that the above can be met, the main operating parameters are defined in 'Operating Rules'. These are issued by the CEBG to the Station Superintendent, and under the Site Licence

requirements, are approved by the Secretary of State for Trade and Industry (DTI). The rules cover many items, such as those concerned directly with operating temperature limitations (fuel can, and graphite), and others covering minimum plant requirements i.e. availability of essential auxiliary electrical systems, trip margin settings and trip minimum requirements. These rules may not be altered unless approved by the DTI. The CEGB is required to set up a Safety Committee for each nuclear plant, and one of the functions of this Committee is to consider rule changes. The membership of the Committee has to be notified to the DTI, and consists of representatives from the plant concerned, the relevant Headquarters' departments of the CEGB, and the U.K. Atomic Energy Authority. The CEGB can impose urgent restrictions, with retrospective reporting to the Safety Committee and DTI.

The principles of operation which define the fuel element can temperature limits are described. The amount of temperature instrumentation provided on the Mk. I GCR plants, although extensive, cannot give the operator complete information on axial and radial temperature distributions; typically channel gas outlet thermocouples are provided on a 1 in 10 basis, with somewhat fewer fuel can thermocouples, and there is very limited information on axial temperatures directly from the reactor. Theoretical assessments are necessary to deduce from reactor information the maximum can surface temperature which has to be related to the limiting temperature deduced from the fault analysis. The actual temperatures existing in a reactor will not correspond exactly with those predicted because of 'random' effects. There is a standard deviation associated with many of the parameters (heat transfer, channel flow, uranium 235 content etc.) and allowance is made for these. Additional points are inherent errors in the accuracy of the measuring equipment, and systematic effects arising for example from the presence of control rods, and absorber bars. It becomes necessary to introduce probability limits into the predictions of actual maximum can temperatures. The criterion adopted is that, following the maximum credible accident tripped on the second line of protection (burst bottom duct coupled with loss of external electrical connections for steel vesselled stations), the probability of any can temperature exceeding the Magnox melting temperature (640°C) is less than 0.01. The calculations demonstrating that this criterion is met involve the transient in each channel for various initial reactor states, the probability of any channel exceeding 640°C and an integration for the whole reactor. Demonstration that the reactors are being run within this criterion involves simple daily assessments, backed up by less frequent, more comprehensive checks. It is of relevance to note that application of this standard, with the siting restrictions imposed, lead to the result that the chance that any member of the public will receive more than 25 rem thyroid dose is extremely remote. Other fault conditions are investigated and are shown to be consistent with this criterion. Different considerations apply for the concrete vesselled stations; in particular the burst bottom duct accident cannot arise, but the same standards for other faults are maintained.

5.2 Operating instructions

Operating Rules define overall limits, and detailed implementation is through Operating Instructions. Under the Site Licence conditions, those operating instructions which have direct relevance to nuclear safety are required to be submitted to the DTI, but do not receive approval. They can be modified by the Licensee, subject to the modifications being consistent with the Operating Rules, but notification of the changes is required within fourteen days.

5.3 Plant maintenance

Plant maintenance is carried out to ensure that the plant performance remains consistent with design criteria. The Site Licence requires the provision of a 'Maintenance Schedule' subject to approval; this schedule parallels the Operating Rules in some respects, in that it lays down the principles of plant maintenance. Included in the schedule are items for which frequent on-load maintenance is necessary (such as guard line testing) and less frequent items which are carried out during the shut down period every two years. The Site Licence also requires a certificate to the DTI to confirm that all maintenance has been completed before approval for continuing operation is given. At the two yearly overhaul period, in addition to the maintenance requirements, a thorough inspection including the pressure circuits of the plants is carried out, and again certification is required.

5.4 Operator training

The importance of operator performance in ensuring safe and efficient operation of the nuclear plants is fully recognised, and great importance is given to operator training. Full scale emergency exercises are regularly held (usually once per year at each plant), supplemented by smaller exercises more frequently. These provide comprehensive simulation of potential accident conditions, and are designed to test both the aspects of immediate plant control and remedial work, together with the wider implications on the public. The exercises are attended by official observers.

6. OPERATIONAL EXPERIENCE

From safety considerations, there have been relatively few items which merit special mention throughout the 88 reactor years operation. Some plant performance aspects of interest are:

6.1 Essential supplies systems

There have been failures of plant forming part of the essential electrical supplies systems. Generally these have only affected small parts of the systems, but there have been three instances when faults on the systems caused loss of supply to important plant. Two of these occurred during periods of severe winter weather. One involved a series of faults on the grid system, resulting in complete isolation of the station. Due to faults on the standby diesel engines, difficulty was experienced in maintaining electrical auxiliary supplies for a short time.

The second incident occurred when, again due to severe weather conditions, a station was disconnected from the grid. In this case the diesel engines were being run as a precaution against loss of grid connection, but due to the operation of protective devices the diesels were lost.

The third occurrence was attributable to a maintenance error, which caused a fault on a switchboard, which in turn affected other parts of the system due to inadequate discrimination in the electrical protection.

In none of these cases was important plant without supply for sufficient time to cause damage, nor was the plant associated with the essential supplies systems damaged significantly.

6.2 Steel oxidation

As reported in [2], oxidation of some steel components in excess of the predicted values has occurred in the Mk. I GCR's. Comprehensive inspections of the reactors have been carried out in the shutdown periods, and these coupled with theoretical assessments have demonstrated that plant safety remains sufficiently high. As precautionary measures, however, additional operational procedures have been implemented to provide continuing reassurance on the safety aspects which in the long term might be affected.

6.3 Fuel element performance

The Magnox fuel elements are normally irradiated for periods in the range three to six years depending on their position in the reactor. Observation of elements removed after their irradiation period showed fin waving in excess of expectation, and subsequent laboratory work has shown that this effect was enhanced by the presence of impurities in the gas coolant, particularly in the higher temperature regions. It was necessary for a limited period at one plant to reduce the maximum fuel can temperature resulting in a small reduction in plant output. The problem has been overcome by very careful control of coolant impurities.

6.4 Bellows cracking

The large gas ducts on the Mk. I GCR's have flexible bellows at various positions; monitoring on certain bellows for CO₂ leakage is carried out. At one station an increase in the CO₂ reading was observed. The circuit was isolated, and subsequent investigation has shown the presence of a small crack between two convolutions. The circuit containing this bellow had

been subject to excessive vibration, and it is believed that the crack was induced by fatigue. It is interesting to note that the failure was very small and did not propagate; this is in accord with predictions for the design adopted.

6.5 Turbine failure

At one of the plants severe turbine disruption occurred following failure of discs during overspeed testing. The significant point of safety interest is that the plant was shut down as expected, and the staff were able to concentrate on the turbine failure.

7. GENERAL CONCLUSIONS

This paper has reviewed some safety aspects of the operational experience of the CEBG with its Mk. I GCR plants, and has described some modifications which have been made. The implications on the design of Mk. II GCR and subsequent reactor systems have been highlighted. Generally, however, the experience has been very satisfactory and no major difficulties have been experienced.

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DESIGN FOR SAFETY OF GAS-COOLED REACTORS

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Abstract—Résumé—Аннотация—Resumen

DESIGN FOR SAFETY OF GAS-COOLED REACTORS.

The important safety characteristics of the Mk II Gas-Cooled Reactors (AGRs) and Mk III Gas-Cooled Reactors (HTRs) are described with particular reference to the safety margins inherent in the casing materials and in the large thermal capacities of the graphite moderators. A few typical fault transients are presented to illustrate these properties. The operating Mk I Gas-Cooled Reactors (Magnox) have given confidence in the predictions of core physics and fault transient models as accurate design tools for gas-cooled reactors. The reactor protection and essential cooling systems are typical of gas-cooled reactors requiring high reliability. The design, construction and testing of these systems for the Hartlepool Mk II GCR station are described. The use of the probability approach to safety is discussed as a useful design tool for synthesizing complex systems. It is demonstrated that reliability assessments of such systems are essential to achieve reliabilities which are appropriate to the consequences of their failure and to give consistency of design between systems. The way of choosing the number of guard lines and trip parameters for reactor protection using a reliability assessment is described. Similarly, the number of pumps, valves and pipework sections for one of the essential cooling water systems, described above, is determined by specifying a target failure rate for the total system. The probability approach is of limited value in dealing with the design of mechanical structures. Whilst the required reliability of structures whose failure could be catastrophic can be determined from a probability criterion, actual failure rates are difficult to substantiate because of lack of relevant statistical data. Instead, agreement between designers and safety assessors is essential to arrive at an acceptable design by applying sound engineering principles, model testing, redundancy of components, monitoring for faults and periodic inspection. An illustration of these principles is given by reference to the design, construction and testing of the prestressed concrete pressure vessel.

ETUDES VISANT A ASSURER LA SECURITE DES REACTEURS A REFRIGERANT GAZEUX.

Les auteurs décrivent les importantes caractéristiques de sécurité des réacteurs à réfrigérant gazeux Mk II (AGR) et des réacteurs à réfrigérant gazeux Mk III (HTR), et mentionnent plus particulièrement les marges de sécurité propres aux matériaux de gainage et aux grosses capacités thermiques des modérateurs au graphite. Quelques transitoires accidentels types sont présentés pour illustrer ces propriétés. Les réacteurs à réfrigérant gazeux Mk I (Magnox) en service ont confirmé les prévisions quant à la physique du noyau et justifié l'emploi de modèles de transitoires accidentels en tant qu'outils d'étude précis pour les réacteurs à réfrigérant gazeux. Les systèmes de protection du réacteur et de refroidissement essentiels sont caractéristiques des réacteurs à réfrigérant gazeux qui exigent une haute fiabilité. Le mémoire décrit l'étude, la construction et l'essai de ces systèmes pour la centrale GCR Mk II de Hartlepool. Les auteurs examinent la méthode de probabilité employée pour la sécurité et mentionnent qu'il s'agit d'un outil d'étude utile servant à synthétiser les systèmes complexes. Ils démontrent que les évaluations de fiabilité de ces systèmes sont essentielles pour réaliser des fiabilités qui soient adaptées aux conséquences de leur défaillance et pour assurer l'uniformité des différents systèmes. Ils décrivent la manière de choisir le nombre de circuits de garde et de paramètres de déclenchement pour la protection des réacteurs en employant une évaluation de fiabilité. De la même manière, ils déterminent le nombre de pompes, de vannes et de sections de tuyauteries pour l'un des systèmes essentiels de refroidissement par eau décrits ci-dessus, en spécifiant un taux de défaillance prévu pour l'ensemble du système. La méthode de probabilité est d'une valeur limitée dans le cas de l'étude des structures mécaniques. Bien qu'il soit possible de déterminer, d'après un critère de probabilité, la fiabilité requise des structures dont la défaillance serait catastrophique, il est difficile d'établir les taux de défaillance effectifs étant donné le manque de statistiques applicables. Au lieu de cela, il est essentiel que les dessinateurs et les

inspecteurs de sécurité s'accordent en vue d'aboutir à une étude acceptable en appliquant des principes solides de construction, en éprouvant les maquettes, en déterminant la redondance des composants, en assurant le contrôle pour la détection éventuelle des défaillances et en prévoyant l'inspection périodique. Pour illustrer ces principes, les auteurs se réfèrent à l'étude, la construction et l'essai de la cuve de pression en béton précontraint.

ВОПРОСЫ БЕЗОПАСНОСТИ ПРИ КОНСТРУИРОВАНИИ РЕАКТОРОВ С ГАЗОВЫМ ТЕПЛОНОСИТЕЛЕМ.

В докладе дается описание важных характеристик безопасности реакторов с газовым теплоносителем Mk II (AGR) и Mk III (HTR), причем особое внимание уделяется запасам прочности материалов для оболочек и высокой теплоемкости графитовых замедлителей. Представлено несколько переходных процессов при типичных неисправностях, с тем чтобы проиллюстрировать эти свойства. Находящиеся в эксплуатации магноксвые реакторы с газовым теплоносителем типа Mk I как точные инструменты конструирования газоохлаждаемых реакторов позволяют делать предсказания в отношении физики активной зоны и моделей переходных процессов при неисправностях. Защита реактора и системы охлаждения типичны для реакторов с газовым теплоносителем, требующих высокой степени надежности. Дается описание конструкции, сооружения и испытания этих систем для атомной станции, состоящей из реакторов с газовым теплоносителем, типа Mk II в Хартлпуле. Использование вероятностного подхода к вопросам безопасности рассматривается как полезный метод конструирования для синтезирования сложных систем. В докладе показано, что оценка надежности таких систем очень важна для достижения той степени надежности, которая необходима для преодоления последствий, возникающих при повреждении систем, а также для обеспечения конструктивной целостности этих систем. Описывается метод выбора числа аварийных линий и параметров аварийных остановок для защиты реактора с помощью метода оценки надежности. Подобным же образом, количество насосов, клапанов и трубопроводов для одной из важных систем водяного охлаждения, упомянутых выше, определяется путем установления плановой скорости развития аварийной ситуации для всей системы. Вероятностный подход малозффективен, когда дело касается конструкции механических устройств. В то время как требуемую надежность устройств, повреждение которых могло бы носить катастрофический характер, можно определить по критерию вероятности, действительные скорости развития аварийной ситуации, вследствие отсутствия соответствующих статистических данных, определить трудно. Поэтому важное значение имеет согласованная работа конструкторов и экспертов по вопросам безопасности для разработки приемлемой конструкции путем применения правильных инженерных принципов, модельных испытаний, контроля аварийных станций и периодических инспекций. Эти принципы иллюстрируются на примере конструирования, сооружения и испытания корпуса, работающего под давлением, из предварительно напряженного бетона.

DISEÑO PARA LA SEGURIDAD DE REACTORES REFRIGERADOS POR GAS.

Se describen en este artículo las importantes características de seguridad de los reactores refrigerados por gas de los tipos MK II (AGR) y MK III (HTR), haciendo especial referencia al margen de seguridad proporcionado por el material de la vaina y por la gran capacidad térmica del moderador (grafito). Se incluyen unos pocos ejemplos típicos de estados transitorios originados por fallos como ilustración de estas propiedades. El funcionamiento de los reactores refrigerados por gas del tipo MK I (Mgnox) ha confirmado la validez de las predicciones de los modelos teóricos para estados transitorios originados por fallos y para la física del núcleo, haciéndolos herramientas de gran precisión para el diseño de reactores refrigerados por gas. La protección del reactor y los sistemas de refrigeración requieren en los reactores refrigerados por gas una gran fiabilidad. Se describen en esta memoria el diseño, la construcción y los ensayos de estos sistemas para la central de Hartlepool del tipo MK II. Se discute el uso de un método probabilístico en cuestiones de seguridad como herramienta de diseño útil en la síntesis de sistemas complejos. Se demuestra que la evaluación de la fiabilidad de tales sistemas es esencial no sólo para obtener una fiabilidad global que sea apropiada a las consecuencias de su fallo, sino también para hacer diseños coherentes de los diversos sistemas. Se describe la forma de elegir los límites de seguridad y los parámetros de disparo para la protección del reactor mediante una estimación de su fiabilidad. Asimismo se determina en este artículo el número de válvulas, bombas y secciones de tuberías para uno de los sistemas esenciales de refrigeración por agua descrito previamente, fijando como dato de partida el ritmo de fallos del sistema global. El método probabilístico es de valor restringido cuando se trata del diseño de estructuras mecánicas. Aunque la fiabilidad requerida en estructuras, cuyo fallo supondría consecuencias catastróficas, podría obtenerse mediante la aplicación de criterios probabilísticos, realmente la falta de información estadística suficiente hace difícil la predicción del ritmo de fallos. En este caso, es fundamental llegar a un acuerdo entre el diseñador y el asesor de seguridad, para obtener un diseño aceptable mediante la aplicación de

principios técnicos seguros, ensayo y verificación de modelos, redundancia de componentes, vigilancia de fallos e inspecciones periódicas. Como ilustración de estos principios se hace referencia al diseño, construcción y prueba de una vasija de presión de hormigón pretensado.

1. INTRODUCTION

In the U.K., the Mk. I Gas Cooled Reactor (Magnox) and Mk. II Gas Cooled Reactor (A.G.R.) stations are based on the concept of containing the active fission products within the fuel cans. The coolant gas circuit is hence devoid of fission product activity in normal operation. Continuous discharge of radioactive effluent arising from the stations are insignificant and this aspect is not a major concern in the safety appraisal of these stations. The double containment of the active products within the cans and the coolant pressure circuit means that safety of the public is vested in the prevention of accidents which would cause both widespread can failures and breach of the pressure circuit. Recognition of the limitations of the canning materials and prevention of these limits being exceeded are important safety characteristics of these reactors. For this reason extensive studies of the transient behaviour of the reactors are carried out during the design stages.

The Mk. III Gas Cooled Reactor (H.T.R.) departs from the "clean circuit" concept of the Mk. I and II Gas Cooled Reactors. However, many of the characteristics of the Mk. III G.C.R. are similar to those of the Mk. II G.C.R.

The reliability to which essential systems should be designed is a subject receiving much attention at the present time. The use of a probability criterion in an attempt to quantify safety design is discussed in section 5. The results of a probability assessment of the reactor safety system and essential cooling systems for the Hartlepool and Heysham Mk. II G.C.R. power stations [1] currently being built by British Nuclear Design & Construction Limited (B.N.D.C.) for the Central Electricity Generating Board (C.E.G.B.) are discussed in Sections 6 and 7.

In all gas cooled reactors the pressure envelope is perhaps the most important single item in guaranteeing public safety. The design, construction and operating principles of the prestressed concrete pressure vessels currently used by Taylor Woodrow Construction for the B.N.D.C. Gas Cooled Reactors are outlined in Section 8.

2. CHARACTERISTICS OF THE MK. II G.C.R. SYSTEM

The temperature of the hottest cans are in the region of 800°C during normal operation at power and although can failures may occur if the temperature is increased above 1100°C, can melting does not occur until a temperature of approximately 1400°C is reached. The graphite moderator is normally maintained at temperatures below 500°C by the re-entrant flow arrangement of the core. This provides a large sink for fission product heat. Fig. 1(a) shows that if a Mk. II G.C.R. is devoid of all coolant flow, the hottest cans do not reach melting point for several hours.

Following a reactor trip, heat is removed from the core by continued gas flow and boiler feed water flow. These post trip flows are approximately 10% of full load flow and are far in excess of the minimum required to remove fission product heating without damaging the fuel. Fig. 1 (b) shows the can temperature transient resulting when 2% gas flow is provided following reactor trip, together with sufficient feed flow (approximately 3%) to maintain the reactor gas inlet temperature near to its full load value. It can be seen that the hottest can temperature does not exceed the can failure temperature of 1100°C and is some 400°C below the melting temperature.

3. CHARACTERISTICS OF THE MK. III G.C.R. SYSTEM

The Mk. III G.C.R. will comprise a prestressed concrete pressure vessel with podded boilers similar to those of the B.N.D.C. Mk. II G.C.R. system. The main differences between the two systems are in the fuel and canning material and the coolant gas.

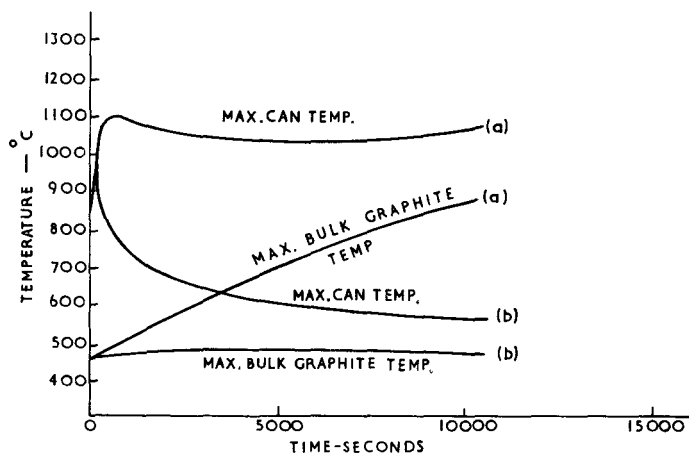


FIG. 1. All circulator run down - Hartlepool.

- (a) To zero core flow
(b) To 2% core flow

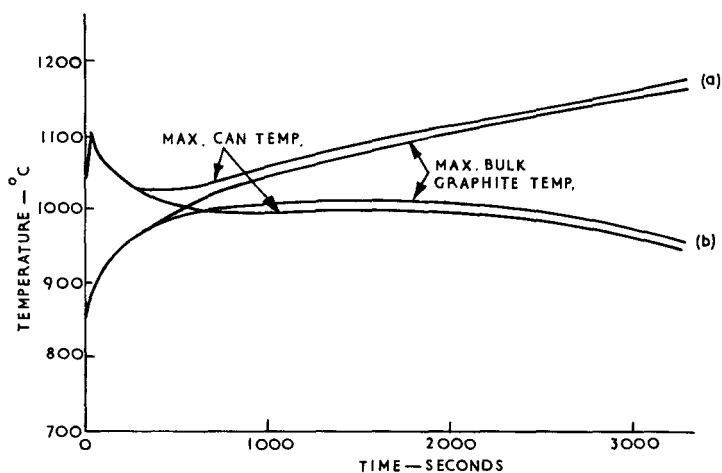


FIG. 2. Rapid reduction of gas flow - Mk III GCR.

- (a) To 1% core flow
(b) To 2% core flow

The Mk. III G.C.R. circuit always contains a small amount of radioactivity. For this reason, and also to conserve the more expensive helium coolant, special design provisions are made, as compared with the Mk. II G.C.R., to reduce gas leakage to a very low level. Also, such gas as escapes can be filtered and discharged from a high stack.

The moderator graphite acts as a very effective sink for fission product heat and for absorbing excess heat produced in the fuel during fault conditions, the heat from the fuel being radiated to the bulk graphite. Fig. 2 (a) shows the effectiveness of the graphite heat sink in a coolant flow stagnation transient following reactor trip and Fig. 2 (b) the effect of 2% core flow after reactor trip.

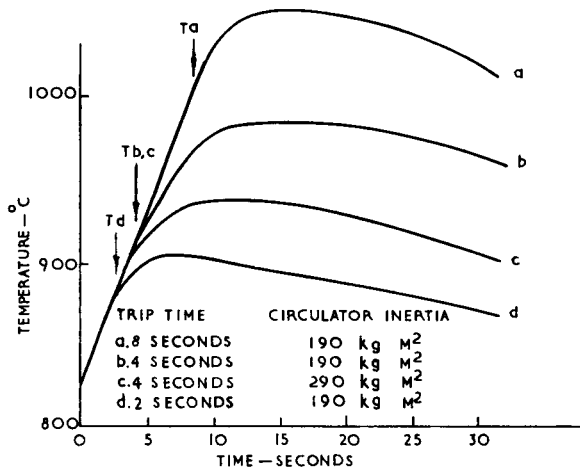


FIG. 3. All circulator run down - Hartlepool.
Maximum can temperature transients.

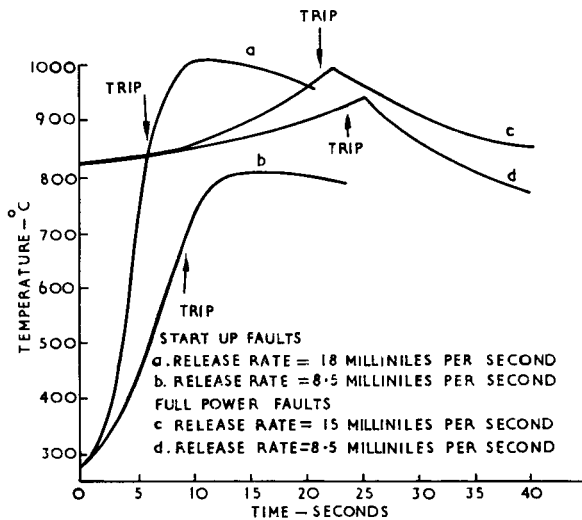


FIG. 4. Reactivity release faults - Hartlepool.
Maximum can temperature transients.

4. FAULT TRANSIENT STUDIES

Reactor kinetic models using analogue or digital computing techniques have been used as a design tool for gas cooled reactors since the early days of the U.K. I.G.C.R.'s. An example of the use of kinetic fault studies to determine the effect of design parameters is typified by an

electrical system fault causing all gas circulators to run down simultaneously. The severity of this transient for the Hartlepool Mk. II G.C.R. system is shown in Fig. 3. The maximum can temperature transient can be reduced by either increasing the circulator inertia from $190 \text{ kg}\cdot\text{m}^2$ to, say, $290 \text{ kg}\cdot\text{m}^2$ (curves b and c) or by providing fault sensors which will trip the reactor within 2 to 5 seconds rather than 5 to 10 seconds. For this fault two prompt tripping parameters are available, namely loss of circulator speed and loss of circulator volts, so the more inconvenient and expensive alternative of increasing circulator inertia is not necessary.

A similar situation arises with postulated control rod withdrawal faults. Fig. 4 (a) shows the effect on the maximum can temperature of different reactivity release rates from a shutdown condition. The transients shown are terminated by the second trip parameter, reactor gas outlet temperature, the first line being neutron flux which gives a much more rapid reactor trip. A similar fault from a full power condition is shown in Fig. 4 (b). In both the start-up and full power cases, the transients resulting from the larger release rates are below the can failure limit but due to the rapidity of the fault, and to allow for some possible deterioration in thermocouple response, the rod speeds are designed to give the lower release rates.

5. SAFETY CRITERIA

It is beyond dispute that systems upon which the safety of the public depend must be extremely reliable. The problem which designers, customers and licensing authorities are faced with, and upon which they must ultimately agree, is – how reliable? Recently, in the U.K. the probability approach to safety has been proposed as a means of obtaining a quantitative method to resolve this problem [2, 3].

The main characteristic of both of these proposed criteria is that the frequency (or probability within a particular period of time) of an accident sequence giving rise to a particular ^{131}I release to atmosphere should decrease as the size of the release increases.

In contrast, the “credible/incredible” criterion states that any accident giving rise to a release of radioactivity above a certain level shall be demonstrated as “incredible”. This criterion is inadequate until “incredible” is defined. If “incredible” is defined as being less than a particular frequency then the “credible/incredible” criterion becomes an extreme case of the probability criterion. The choice of criterion is outside the terms of reference of the designer, it being a matter for safety advisory bodies and licensing authorities to decide upon. However the designer and customer would be in a better position to know how much redundancy to build into their systems if a quantitative criterion was adopted and, as a result, failure probabilities specified for particular system functions.

For systems involving many components, such as electrical and electronic logic circuits, and cooling systems comprising a network of pipes valves and pumps, a target failure rate for the system is an essential requirement before a designer can arrive at a system in which he has any confidence.

At the request of the C.E.G.B., B.N.D.C. are carrying out the safety assessment of the Hartlepool and Heysham stations using a probability criterion. From this criterion maximum allowable probabilities of failure to operate have been derived for each system involved in the safety of the station and the systems have been analysed to ensure these failure rates are not exceeded. For the safety system a probability of failure of 1 in 10^7 demands is required. The essential cooling system failure probabilities are in general somewhat higher due to the diversity of cooling facilities.

6. REACTOR SAFETY SYSTEM

Reactor trip parameters are chosen using transient studies of the reactor and boilers as indicated in Section 4. The redundancy of transducers, amplifiers, relays, contractors and segregated cable routes to effect reactor tripping has previously been based on simple, but sound, rules and the use of high quality components.

For the Hartlepool and Heysham safety systems, a four guard lines system is used. The control rods fall into the reactor if any two of the four guard lines become de-energised. The guard lines comprise logic units appropriate to each trip parameter. The logic units are made up of solid state "laddic"¹ modules rather than the more usual relay and contact circuits.

The trip parameters for each type of fault are listed in Table I together with the probability of failure to trip for each parameter. The latter have been derived using equipment failure rates based on data collected from industry. These rates are pessimistic since the equipment used in the safety system is usually of a higher quality than that generally used in industry and to which most data applies. It can be seen that the combined "theoretical" failure probabilities of the two lines of protection for each fault easily meet the specified rate of 1 in 10^7 demands.

It is clearly impossible to prove that a system can achieve a failure probability as low as 1 in 10^7 . Indeed, it is impracticable to perform enough tests on an installed system to demonstrate a rate very much greater than 1 in 10^2 . It would be necessary to invoke 300 demands without any failures on the system to give 95% confidence that the failure rate is less than 1 in 10^2 . The best that can be done is to design such systems using high quality components and as much physical segregation between parts of the system as is practicable. Also, at all stages of construction, the system should be inspected for qualities of material and workmanship; finally, regular testing and maintenance of the equipment during the commissioning and the operating periods must be carried out.

7. ESSENTIAL COOLING SYSTEM

The similarity between the Mk. II G.C.R. and Mk. III G.C.R. systems is such that the post trip cooling requirements are almost identical.

In the Hartlepool and Heysham reactors, the gas circuit is arranged so that natural circulation of coolant gas will occur should the gas circulator auxiliary motors fail to start, provided the gas circuit is pressurised. Hence the essential function following a trip is to provide adequate feed to the boilers. This is normally provided by four essential feed water pumps. The pumps are dependent on electrical supplies and a cooling water supply for continued operation. The latter is provided by the turbine ancillaries cooling water system and its heat is rejected to the sea through the essential sea water system. These essential systems are shown in Fig. 5. The essential systems involved in pressurised reactor faults are shown in the fault sequence diagrams in Fig. 6.

At the Hartlepool station, the essential sea water and turbine ancillaries cooling water systems operate continuously. Each of these systems is divided into two halves by normally closed valves. This is to prevent a pipework failure in one half of the system from affecting the whole whilst at the same time allowing water to be pumped from one half to the other if the pumps on one half system fail. This arrangement leads to two pumps on each half system so that any one pump can be maintained with the reactors at power. The power supplies to the four pumps in each essential system are provided from separate supply boards, normally energised by the grid or the main turbo-alternators through the four unit transformers. In the event of loss of grid and main turboalternator supplies the power is derived from four separate gas turbines. Fault sequences for the normally running systems fall into two categories, those to determine the frequency of complete loss of the system such that a reactor trip would occur with no subsequent cooling capability, and those to determine the probability of complete system failure in the period during which they are essential after a reactor shutdown or trip.

Several interesting points have resulted from this analysis. The most obvious observation was that the failure rate for the system was very dependent on the failure rate ascribed to fractures of valve casings. Data was available for all types of valve faults but not for the break down of the types of faults. A conservative value of 2×10^{-3} per valve per year was taken to see what effect

¹ "Laddic" modules are manufactured by G.E.C.-Elliot Process Automation Ltd, New Parks, Leicester, United Kingdom.

TABLE I
HARTLEPOOL REACTOR TRIP PARAMETERS

Fault	1st Trip Parameter	Probability of Failure	2nd Trip Parameter	Probability of Failure	Total Probability of Failure
Control rod group run out	Neutron flux	1.1×10^{-3}	Channel Gas Outlet Temperature	3.7×10^{-4}	$< 10^{-7}$
All circulator rundown	Circulator speed	1×10^{-6}	Circulator Voltage	3.2×10^{-4}	$\ll 10^{-7}$
Loss of boiler feed flow	Feedflow 'X'	2.4×10^{-5}	Feedflow 'Y'	2.4×10^{-5}	$\ll 10^{-7}$
Loss of vessel pressure	Gas pressure	$< 10^{-7}$	Channel Gas Outlet Temperature	3.7×10^{-4}	$\ll 10^{-7}$
Circulator inlet guide cane	Inlet Guide Vanes position	$\ll 10^{-7}$	Channel Gas Outlet Temperature	3.7×10^{-4}	$\ll 10^{-7}$

this had on the analysis. Due to the large number of valves in the system and the fact that a burst could lead to loss of two of the pumps in one half of the pump house, the failure rate for the whole system was found to be approximately 10^{-5} per year. This has been reduced to 10^{-6} per year by the inclusion of an interconnection with the main cooling water system. The assumed valve failure rate is thought to be higher than is the case in practice. Using the same data, the sea water system failure probability in a ten hour period following a reactor trip was found to be extremely low, less than 1 in 10^8 .

The failure rate of 10^{-6} per year for the sea water system is extremely low. However, complete loss of post trip cooling for a significant period following trip could ultimately lead to a large fission product release to the atmosphere. On the assumption that perhaps 10% of the total reactor ^{131}I inventory might be released, the system failure rate is at least a factor of ten too high.

A theoretical system failure rate as low as 1 in 10^6 years raises the question whether other factors are not more likely. For example, partial blockage of sea water intake screens has occurred at several stations. Although the essential sea water flow for post trip duty is only a few percent of the total sea water intake during normal operation, is the frequency of total blockage less than 1 in 10^6 years? Also, system damage could occur due to missiles from other plant failures. Are these less frequent than 1 in 10^6 years? Since these questions cannot be answered a possible solution is to provide a separate source of cooling water flow to the emergency boiler feed pumps and the pressure vessel cooling water system. This has been included in the Hartlepool and Heysham stations by providing diesel driven pumps, supplied from a public water reservoir. In this way the total failure rate of cooling to the reactors is made extremely low and has no difficulty in meeting the probability criterion.

8. PRESTRESSED CONCRETE PRESSURE VESSELS

As the unit size of gas cooled reactors became larger, the need for increased dimensions and pressures, and the desirability of integral designs in which the entire gas circuit was housed in a single pressure envelope led to the development and construction of prestressed concrete pressure vessels. Experience in the form of the progressive development of size and style was not available, and yet the requirement for the unquestionable safety of the prestressed concrete pressure vessel was paramount.

The fulfilment of this requirement is based upon the following criteria which have been satisfied by research and development programmes:

- (i) That the structural model of failure of the vessel is known and takes the form of the ductile fracture of prestressing tendons, thus giving a steady progression to failure, and a predictable minimum failing pressure.
- (ii) That the steel liner remains gas-tight to well beyond the design pressure.
- (iii) That the relative movements of different parts of the pressure vessel structure over the operating life of the reactor can be predicted sufficiently accurately to permit the satisfactory design and operation of reactor components supported by, or passing through, the pressure vessel.
- (iv) That the component materials and techniques employed in the construction are well proven, and capable of satisfactory inspection and control.
- (v) That the adequacy of both the design and the construction can be satisfactorily confirmed by testing prior to commissioning of the vessel, and also the subsequent adequacy of the vessel during operation, confirmed by appropriate surveillance procedures.

The way in which these criteria have been fulfilled in the design and construction of the Hartlepool vessels will now be briefly examined:

- (i) The ultimate pressure of the concrete vessel structure is taken as that which would cause structural failure, either by primary crushing of the concrete, by rupture of the prestressing tendons, or by the intervention of a mechanism leading to local shear or structural collapse. The preferred mode is by rupture of the tendons, which results in an effectively ductile failure of the vessel.

The vessel structure has been analysed for all conceivable failure modes to determine its minimum ultimate pressure. In the weakest mode, the pile cap remains intact and rigid, and plastic hinges are formed at the upper and lower haunches, and at the equator of the barrel, with corresponding vertical cracks opposite the boiler penetrations. This failure mode has been verified by observation of a number of concrete pressure vessel models, including the 1/10th scale model of the Hartlepool vessels.

The criteria adopted in the analysis is conservatively chosen, so as to yield a "lower bound" failing pressure. The analysis shows that failure occurs by rupture of the equatorial hoop prestressing tendons which it is assumed occurs when these reach a strain of 1%. The 1/10th scale concrete model was tested to 3 x design pressure, when, although substantial cracking had occurred, the model was structurally intact, and the hoop strain in the equatorial prestress was found to be 0.7%.

The pile cap of the vessel is heavily penetrated by stand-tubes, the configuration of which excludes effective quantities of prestressing tendons or bonded reinforcement. All prestress is thus applied by hoop tendons around the periphery of the pile cap. The analytical problem presented is thus extremely difficult, and the ultimate strength of the pile cap has been verified in a parametric study of 20 concrete models representing the upper part of the vessel to a scale of 1/24th. This has shown that the pile cap has an ultimate strength in excess of 4 x design pressure, and justifies the assumptions made in the ultimate load analysis of the concrete vessel.

- (ii) The design pressure specified for the vessel is a maximum value which takes account of those accident conditions which are identified for the reactor system. Nevertheless, the designer has to assess the overload characteristics of the vessel to satisfy himself that its mode of failure is acceptable. This is done, as described above, by checking that the vessel, in all its parts, can accept a hypothetical pressure of 2.5 times design pressure. It is assumed, in this calculation, that the liner remains intact and continues to transmit gas pressure to the concrete structure.

In the design of the Hartlepool and Wylfa vessels, an assessment was also made of the response of these structures under various assumed conditions of liner leakage.

The vessel liner is fabricated in ductile steel and to very high standards of workmanship. The tensile strain in the liner is low, since it is constrained to deform with the concrete in which tensile stresses are limited to 400 lbf/in². Even at the postulated "lower bound" failing pressure of the concrete vessel structure (2.5 times design pressure), the maximum imposed tensile strain in the liner should be less than 2,000 microstrains. Therefore, the likelihood of significant liner leakage is remote, even after prolonged irradiation.

- (iii) Since concrete creeps under the influence of stress and temperature, the prediction of the progressive deformation of the concrete vessel during operation influences the design of other reactor plant and components.

Dynamic relaxation techniques have been used for the elastic analysis of the vessel. Good correlation was obtained between strain measurements on a 1/80th scale aluminium filled epoxy resin model of the vessel, on the 1/10th scale concrete model and the analytical predictions.

The creep behaviour of the actual concrete used in the construction has been measured at three temperatures, in the context of a general creep programme carried out over the last eight years, results of which have been used to predict the creep of the vessel over its operational life. Multi-axial effects have been represented in the analysis by adopting the principle of superposition, which it is believed gives acceptably accurate limiting values.

Absolute and relative movements at significant points in the vessel structure have been calculated to give ranges of minimum and maximum expected values at various stages in the life of the vessel. Each calculated range has been based upon individually assessed variations of concrete specific strain, prestress, structural rigidity, shrinkage and coefficient of expansion.

- (iv) The steel liner is selected with an impact value of 30 ft. lbs at -10°C and a yield range of 15-20 tons/in². All welds are fully radio-graphed to ensure acceptability, except that in

locations where radiography is not possible a full magnetic crack detection survey is made. The concrete mix uses locally available materials, selected after careful survey, and is designed to give satisfactory values of strength, workability, creep, shrinkage, heat of hydration and expansion coefficient.

Comprehensive quality control procedures are implemented in the construction, and standard deviations of 380-390 lbf/in² on a mean 28 day cube strength of 8,030 lbf/in² are being achieved. Careful attention is given to obtaining full compaction by vibration and the time between adjacent concrete pours is regulated to minimise the effects of differential cooling strains resulting from heat of hydration.

Hoop prestressing tendons are formed by winding 33 layers of 0.2 in diameter wire into channels formed in the vessel walls, at a tension of 70% of the guaranteed ultimate tensile strength of the wire. A specially designed wire winding machine employs electro-hydraulic control systems to control wire tension to +2%, and the lay of the wire to precise geometric location.

Vertical prestressing tendons each consist of twenty eight 0.7 in. diameter "Dyformed" strands with the Cable Covers Ltd. anchorage system. The tendon has been developed and tested at full scale in a 2,000 ton capacity rig, by loading some 30 tendons to failure.

- (v) Concrete vessels are normally proof pressure tested at ambient temperature to 1.1 x design pressure. Observation of a considerable number of strain gauges cast in the concrete and attached to the vessel liner enable comparison with analytical predictions. The correlation obtained in the proof pressure test of the Wylfa vessels was very good, and superior to previous correlations between analytical prediction and strains measured in concrete models. Thus the proof pressure test is used to give assurance that the vessel behaves substantially in the manner intended by the designer.

Surveillance of the vessel during operation can be satisfactorily accomplished by: visual examinations of the vessel surface and exposed prestressing components; measurement of force in prestressing tendons; removal of prestressing tendons for visual examination; readings of embedded concrete strain gauges (these gauges do not have an indefinite "life" but should be available during the early life of the reactor).

These measures enable verification to be made of the continued acceptability of the vessel during the operating life of the reactor.

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EVOLUTION DES IDEES FRANÇAISES SUR LA SURETE DES REACTEURS

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Abstract—Résumé—Аннотация—Resumen

EVOLUTION OF FRENCH IDEAS ON REACTOR SAFETY.

Until recently the "maximum hypothetical accident" method was universally used. While waiting for a quantitative probability method to be developed to the point where it can be used generally, we have chosen an analytical method of safety evaluation. This method allows safety measures to be selected objectively and facilitates a permanent exchange of ideas at each stage between the design, construction and operation teams and those responsible for safety; it is based on the continuous analysis of the availability and the potential accident behaviour of each barrier or shield between dangerous products and the public. The review is quantitative when possible and qualitative when this is not the case.

For each type of reactor, three or four barriers are provided which are both tight and resistant: cladding, primary circuit, primary containment and, possibly, secondary containment. Each analysis must bring out the progressive character of safety that can be summarized in three interdependent aspects: prevention-barrier validity condition, limits and safety margin; detection; and safety action to prevent or limit fission-product release and allow restoration of the failed barrier. The behaviour of each barrier is studied in three stages: normal operating conditions; normal transient conditions; and accident transients. To synthesize the evaluation of the barriers and to study their interdependence, descriptions of large hypothetical accidents are drawn up; a cross-check with probability method data is attempted on each possible point. An example of this application is given for the safety circuits of the fast reactor PHENIX.

The chief topics of interest are dealt with for the different types of reactors: for water-cooled reactors — containment and pressure-vessel integrity, emergency cooling; for fast reactors — fuel and cladding failures, molten fuel interaction with sodium, sodium boiling, sodium-water reactions, sodium fires; for graphite-gas reactors — out-of-pile firing for fire studies, in-pile test of failed fuel. A particular effort is applied to: (1) Mechanical studies — design and construction techniques for large steel pressure vessels, prestressed concrete vessels, internal structures; Rationalization of architecture and mechanisms; Material stress calculations under normal or accident conditions. (2) Radiological safety studies — Radionuclide transfer through primary sodium to argon blanket; Fission-product diffusion analysis; Atmospheric study techniques with tracers. Essential but expensive, safety studies are an area in which international co-operation should play an important part; work carried out by the CEA in collaboration with organizations abroad is promising in this regard.

EVOLUTION DES IDEES FRANÇAISES SUR LA SURETE DES REACTEURS.

Jusqu'à ces dernières années, la méthode de l'accident maximal prévisible a été universellement employée; tant qu'une méthode quantitative du type de la méthode des probabilités n'a pas été améliorée suffisamment pour devenir d'un emploi général, on adopte un procédé analytique d'examen de sûreté. Cette méthode permet une explication des choix de sûreté et un dialogue permanent entre intéressés; elle comporte l'examen continu de chaque barrière entre produits dangereux et extérieur et de son comportement en cas d'accident. L'analyse est quantitative lorsque c'est possible, sinon, elle est qualitative.

Pour chaque type de réacteur les barrières, étanches et résistantes, sont au nombre de trois ou quatre: gaine, circuit primaire, confinement primaire et éventuellement confinement secondaire. Toute analyse doit faire ressortir le caractère progressif de la sûreté, qui peut se résumer en trois étapes non indépendantes: prévention (conditions de validité de la barrière, limites, marges de sûreté), surveillance, action de sécurité (en vue de prévenir ou limiter l'émission des produits de fission, et de restaurer éventuellement la barrière défaillante). L'examen de chaque barrière est effectué sous trois aspects: conditions normales, transitoires normales, transitoires accidentels. Pour synthétiser cet examen des barrières et étudier leur interdépendance, on décrit d'importants accidents hypothétiques: des recouplements sont effectués avec la méthode probabiliste. Un exemple d'application sur PHENIX est donné.

Les études de sûreté sont centrées sur différents points suivant les types de réacteurs: pour les réacteurs à eau: intégrité des cuves de pression, refroidissement de secours; pour les réacteurs rapides: défaillances du combustible et des gaines, interaction sodium-combustible fondu, ébullition du sodium, réactions sodium-eau, incendies de sodium; pour les réacteurs graphite-gaz: boucle hors pile pour étude des incendies, comportement en pile du combustible dans des gaines percées. Deux domaines sont particulièrement étudiés: 1) la mécanique (techniques de construction de grandes cuves en acier, caissons en béton précontraint, structures internes; rationalisation des architectures et mécanismes; calcul des sollicitations des matériaux en régime normal et d'accident) et 2) la sûreté radiobiologique (transferts de radionucléides à travers le sodium vers l'argon, diffusion des produits de fission, études atmosphériques par traceurs). Coûteuses mais indispensables, les études de sûreté sont un domaine où la collaboration internationale est essentielle, d'où les travaux de coopération entrepris par le CEA avec des organismes étrangers.

РАЗВИТИЕ ИДЕЙ, КАСАЮЩИХСЯ ОБЕСПЕЧЕНИЯ БЕЗОПАСНОСТИ РЕАКТОРОВ, ВО ФРАНЦИИ.

До последнего времени при оценке безопасности повсеместно применяли метод оценки по максимально опасной возможной аварии. В ожидании усовершенствования количественного метода (типа метода вероятности, разработанного в Англии для получения надежных данных) во Франции был принят аналитический метод испытаний на безопасность. Этот метод представляет возможность выбора способа обеспечить безопасность и постоянного обмена мнениями между заинтересованными лицами: проектировщиками, строителями, эксплуатационниками и ответственными за безопасность. Метод заключается в непрерывном наблюдении за наличием аварийного состояния на каждой ступени защиты и за функционированием каждого барьера между опасными продуктами и внешней средой. По возможности применяют количественный анализ, а в остальных случаях — качественный, если дело не касается активной зоны.

Для каждого типа реактора существуют три или четыре барьера, обеспечивающие герметичность и надежное средство защиты: оболочка твэла, первый контур, первая защитная оболочка реактора и, возможно, вторая защитная оболочка. Любой анализ должен выявлять прогрессирующий характер безопасности, который можно свести к трем независимым этапам: предотвращение — создание надежных средств защиты, выбор пределов, зоны безопасности; обнаружение; меры безопасности, направленные на предупреждение или ограничение утечки продуктов деления и на восстановление защитной преграды. Наблюдение за каждым средством защиты проводят в трех аспектах: нормальные рабочие условия; нормальные переходные процессы; и аварийные переходные процессы. Чтобы оценить эффективность различных средств защиты и изучить их взаимозависимость, описаны возможные аварийные ситуации: проверка фактов путем сопоставления осуществлена методом вероятности. Пример такого применения приводится для системы защиты реактора на быстрых нейтронах PHENIX.

Наиболее важные моменты при оценке безопасности различны в зависимости от типа реактора: для водяных реакторов — целостность корпуса высокого давления и защитной оболочки, резервное охлаждение; для быстрых реакторов — разрушение топлива и оболочки, взаимодействие натрия с расплавленным горючим, кипение натрия, реакция натрия с водой, горение натрия; для газо-графитовых реакторов — вне реакторные петли для изучения самовозгорания, поведение горючего в топливной сборке в поврежденных оболочках. Наиболее подробно изучены две области: (1) механика — проектирование и изготовление больших стальных корпусов; корпусов давления из предварительно напряженного бетона, внутренние конструкции; рационализация конструкции и механизмов; расчет напряжений материалов в нормальном (вибрации) и аварийном режиме (землетрясения, взрывы, пожар); (2) радиология — проникновение радиоизотопов из натрия в аргон; анализ распространения продуктов деления; исследование атмосферы с помощью индикаторов. Несмотря на то, что испытания на безопасность весьма дороги, они должны проводиться в рамках международного сотрудничества, примером чего является сотрудничество Комиссариата по атомной энергии Франции с зарубежными организациями.

EVOLUCION DE LAS IDEAS FRANCESAS SOBRE LA SEGURIDAD DE LOS REACTORES.

Hasta estos últimos años se utilizaba universalmente el método del accidente máximo previsible; puesto que aún no se dispone de un método cuantitativo, del tipo del método de las probabilidades, suficientemente perfeccionado para su utilización de forma general, nosotros hemos adoptado como examen de seguridad un procedimiento analítico. Este método permite una manifestación explícita de las variables seleccionables en seguridad y un diálogo permanente entre los interesados; lleva consigo el examen continuo del comportamiento, en caso de accidente, de cada barrera entre los productos peligrosos y el exterior. El análisis es cuantitativo siempre que sea posible y cualitativo en otro caso.

Para cada tipo de reactor existen tres o cuatro barreras estancas y resistentes: vaina, circuito primario, confinamiento primario y, en su caso, confinamiento secundario. Todo análisis debe resaltar el carácter progresivo de la seguridad que puede resumirse en tres etapas no independientes: prevención (condiciones de validez de la barrera, límites, márgenes de seguridad), vigilancia, y acción de seguridad (con objeto de evitar o limitar la emisión de los productos de fisión y restaurar, si es posible, la barrera amenazada). Se hace un examen de cada barrera según tres aspectos: condiciones normales, estados transitorios normales y estados transitorios por accidente. Para hacer la síntesis de este examen de las barreras y estudiar su interdependencia se describen importantes accidentes hipotéticos: se hacen retoques con el método probabilístico. Se da un ejemplo de aplicación con el reactor PHENIX.

Los estudios de seguridad son diferentes según los tipos de reactores: para los reactores de agua: integridad de las vasijas de presión, refrigeración de emergencia; para los reactores rápidos: fallos del combustible y de la vaina, interacción sodio-combustible fundido, ebullición del sodio, reacciones sodio-agua, incendios de sodio; para los reactores grafito-gas: circuito fuera del reactor para estudios del incendio, comportamiento en el reactor del combustible con vainas perforadas. Se estudian de modo especial dos campos: 1) la mecánica (técnicas de construcción de grandes vasijas de acero, recipientes de hormigón pretensado, estructuras internas; racionalización de estudios y mecanismos; cálculo de tensiones en materiales en régimen normal y de accidente) y 2) la seguridad radiobiológica (transferencia de radionucleidos del sodio al argón, difusión de los productos de fisión, estudios atmosféricos). Los estudios de seguridad, indispensables aunque costosos, pertenecen a un campo en el que la colaboración internacional es esencial, por lo cual el CEA ha emprendido trabajos en colaboración con organismos extranjeros.

GENERALITES

Jusqu'aux années récentes, les incertitudes concernant certains phénomènes ou données physiques et certains aspects technologiques liés à la sûreté des réacteurs ont conduit à des analyses de sûreté où l'on cherchait à parer à une grande diversité d'accidents possibles par la prise en compte d'un « accident enveloppe », ou « accident maximal prévisible », choisi avec un grand pessimisme. Cette méthode certainement arbitraire a conduit à des dispositions apparemment raisonnables, mais elle ne pouvait conduire à une optimisation globale, économique et humaine, des installations nucléaires.

Les progrès réalisés dans la connaissance des phénomènes permettent aujourd'hui une approche plus rationnelle et une conception quantitative dont l'exemple le plus caractéristique est celui de la méthode anglaise des probabilités. Cette méthode présente le grand intérêt de fournir des conclusions quantitatives à la fois sur les effets et sur les origines des diverses chaînes accidentelles et c'est certainement à moyen terme une approche de ce type qu'il sera souhaitable d'adopter. Mais son emploi est actuellement limité sur trois plans: les données concernant la fiabilité réelle des divers composants sont encore incomplètes, notamment pour les gros matériels; la connaissance de certains processus physiques accidentels est insuffisante; il est difficile de décider quel risque on peut ou non accepter pour les populations compte tenu du contexte psychologique des dangers nucléaires dans l'opinion publique.

Transitoirement nous adoptons une méthode analytique d'examen permettant de faire ressortir aussi objectivement que possible les choix de sûreté plus directement accessibles aux projeteurs, aux constructeurs et aux exploitants, par un dialogue permanent avec les organismes de sûreté. Cette méthode comporte l'examen continu de la disponibilité et du comportement en cas d'accident de chaque « barrière » séparant les produits dangereux de l'extérieur. L'analyse de la disponibilité est quantitative lorsque cela

est possible, qualitative dans les autres cas en tenant compte au mieux de l'état des connaissances. Parallèlement, les études de sûreté se concentrent sur les aspects dont les incertitudes imposent aujourd'hui une analyse qualitative.

METHODE D'ANALYSE ADOPTEE

La protection des populations contre les conséquences d'un relâchement accidentel de produits de fission repose sur l'interposition en série de «barrières» étanches entre le cœur du réacteur et le public. L'analyse de sûreté consiste alors à s'assurer de leur validité et de leur fonctionnement correct dans les conditions normales ou en cas d'accident.

L'analyse de sûreté de chaque barrière est ensuite menée de manière progressive. Pour chaque condition de fonctionnement on distingue trois actions successives mais non indépendantes:

- La prévention: la validité de chaque barrière doit être démontrée par le choix des matériaux qui la composent, par l'adaptation de ces matériaux aux conditions de fonctionnement et par le maintien rigoureux dans le temps des caractéristiques imposées ou des marges requises. Il est essentiel d'autre part de bien mettre en évidence les limites technologiques, de manière à définir en toute connaissance de cause les marges entre les conditions de fonctionnement et ces limites.
- La surveillance, destinée à détecter toute entrée à l'intérieur de la marge précédemment définie afin d'être en mesure dans ce cas de déclencher à temps une action correctrice automatique ou manuelle pour revenir aux conditions normales.
- L'action de sécurité, qui a pour objectif, en cas de dépassement accidentel des limites technologiques, d'effectuer une correction en vue de prévenir l'émission des produits de fission ou d'en limiter l'ampleur et de permettre une reprise du fonctionnement normal après restauration éventuelle de la barrière défaillante.

L'examen de ces trois actions est mené pour trois types principaux de fonctionnement:

- Les conditions de fonctionnement normal, cas le plus simple et le mieux défini, qui, après détermination des limites technologiques de la barrière et des incertitudes qui demeurent, aboutit à la fixation de caractéristiques ayant des conséquences économiques majeures, comme celle de la fixation de la puissance maximale de l'installation. Tout progrès dans les connaissances peut amener ensuite à une réduction des marges.
- Les transitoires normaux de fonctionnement (démarrages, montées en puissance, variations de charge): leur examen doit être spécialement attentif (ce point a quelquefois été négligé dans le passé et a conduit à des accidents sérieux), et l'on doit s'efforcer de définir la marge du fonctionnement permanent de telle sorte qu'elle absorbe les transitoires normaux sans qu'il soit nécessaire de déclencher les actions correctrices.
- Les transitoires accidentels provoqués par des défaillances du matériel ou des erreurs humaines: l'examen des divers cas est délicat et l'on est souvent conduit, à ce niveau, à des analyses qualitatives. L'établissement des diverses chaînes accidentelles permet de sélectionner les points critiques pour en améliorer la fiabilité ou la surveillance, de choisir les types de parades et d'orienter le cas échéant des études spéciales.

Pour chaque type de réacteur, les barrières réputées à la fois étanches et résistantes sont au nombre de trois ou quatre: gaine, circuit primaire, confinement primaire et éventuellement confinement secondaire. Chacune d'elles est séparément examinée en détail suivant les modalités qui viennent d'être exposées.

Pour synthétiser l'examen des barrières ainsi effectué et pour évaluer en particulier leur indépendance les unes par rapport aux autres, ce qui est essentiel dans un bilan de sûreté, une description d'accidents-types de grande ampleur est ensuite établie. Cette démarche finale garde à notre sens un certain caractère conventionnel, mais elle a le mérite de permettre une estimation du comportement dynamique des produits dangereux au cours de leur transport possible depuis le cœur jusqu'à l'enceinte extérieure et de fournir ainsi un ordre de grandeur des conséquences radiologiques pour le site, si l'intégrité de toutes les barrières devait être atteinte.

Nous avons mentionné qu'un recoupement avec la méthode probabiliste est tenté en cours d'analyse chaque fois que cela est possible. Un exemple d'application est donné ci-dessous pour les circuits de sécurité de la centrale à neutrons rapides PHENIX.

L'étude effectuée dans cette centrale était destinée à montrer les avantages de l'analyse de sûreté par les méthodes probabilistes. L'exemple choisi est la détermination de la probabilité d'arrêt du réacteur par chute de barres à la suite d'une interruption du réseau supérieure à 3 s. Un code de calcul écrit par l'UKAEA et adapté au calculateur IBM 360 est utilisé (code NOTED).

Les chaînes d'arrêt du réacteur peuvent être déclenchées soit par le signal choisi (intitulé « mini-tension 3 s », voir figure 1), soit par d'autres signaux: arrêt de plus d'une pompe à sodium primaire, rapport débit/puissance, augmentation des températures du cœur, et enfin arrêt sur intervention de l'opérateur. Les barres de contrôle sont réparties en deux groupes de trois, indépendants l'un de l'autre. On évalue à deux le nombre de barres nécessaires à l'arrêt du réacteur.

Dans le code utilisé, un organigramme décrit la structure de ces chaînes de sécurité; le code évalue la probabilité d'arriver à END (chute d'au moins deux barres) en partant du signal d'entrée choisi.

Les différents postes d'arrêt ne fonctionneront pas simultanément. C'est l'un des mérites du code utilisé que de faire intervenir ces décalages en temps. L'arrêt des circuits d'alimentation des barres, jouant dans le sens de la sécurité, n'est pas pris en compte.

Actuellement, l'analyse des circuits est terminée; il reste à entrer, suivant les formats du code, la description de l'organigramme et les valeurs des retards et des fiabilités.

Les valeurs de fiabilité de certains composants étant assez mal connues, une étude paramétrique sera parfois nécessaire. Là encore, les indications données par le code sur les probabilités partielles (par sous-ensemble) éviteront de paramétrer les composants peu déterminants.

Les précautions prises dans l'élaboration de la chaîne de sécurité de PHENIX aboutiront très probablement à une fiabilité résultante excellente. Cette analyse sera alors reprise pour vérifier que cette fiabilité n'entraîne pas en contre-partie un taux excessif de chutes intempestives. Une telle double analyse devrait contribuer à l'optimisation économique des sécurités.

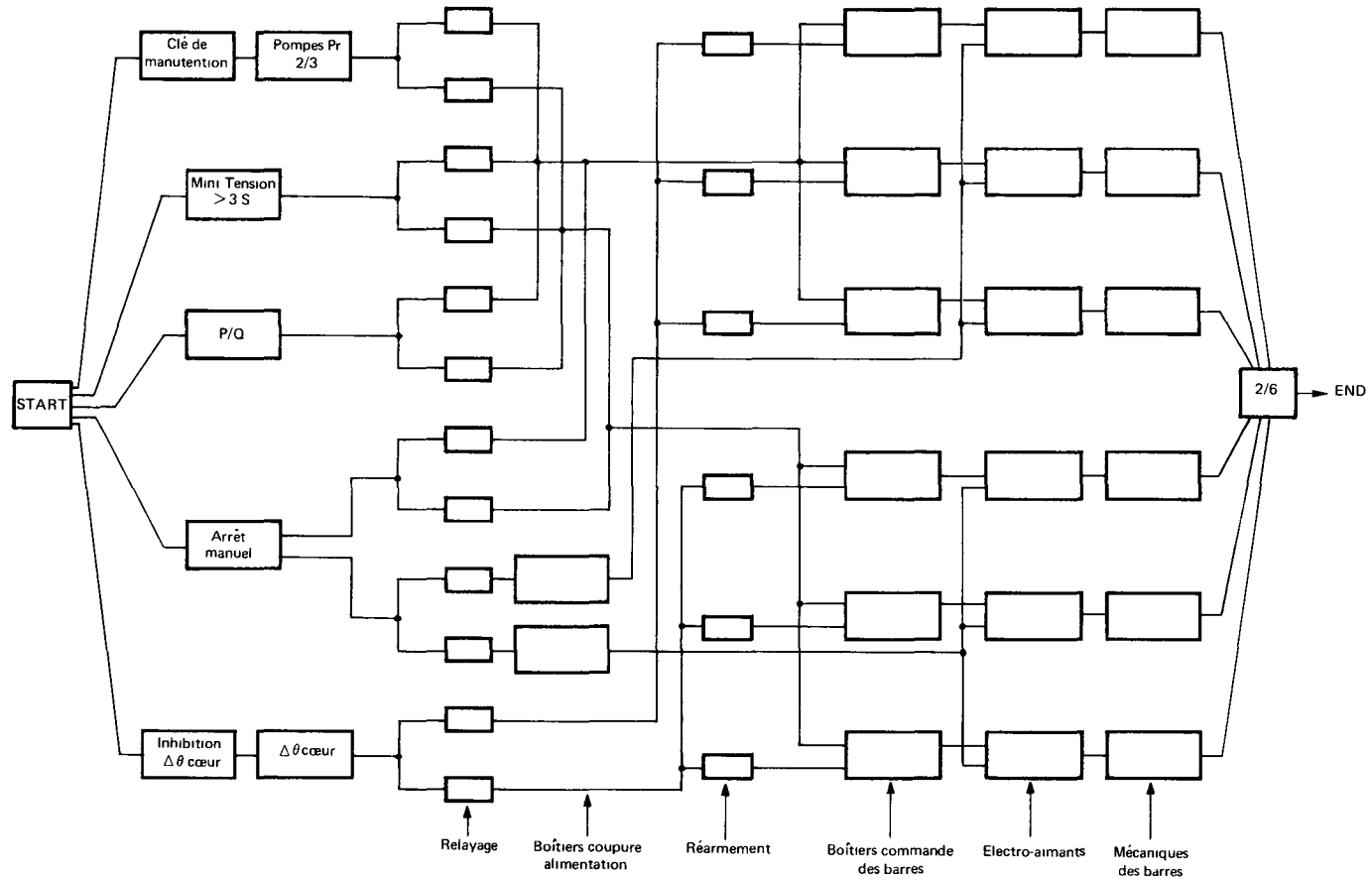


FIG.1. Code NOTED: Structure des chaînes de sécurité de PHENIX.

LES POINTS D'INTERET DE NOS ETUDES DE SURETE

Les études de sûreté sont centrées sur les sujets fondamentaux dont la compréhension est indispensable pour pouvoir appliquer une méthode quantitative et réaliser un meilleur compromis sûreté-économie.

Les études liées à la sûreté des réacteurs à eau sont en cours de démarrage dans notre pays. Elles complètent les études étrangères et concernent: l'intégrité des cuves de pression (conception, fabrication, contrôles et surveillance en fonctionnement); le refroidissement de secours en cas de perte de réfrigérant primaire (aspects mécaniques et thermo-hydrauliques); le comportement de l'intégrité de l'enceinte de confinement dans les situations normales et anormales.

Des recherches théoriques et expérimentales importantes sont menées sur la sûreté des réacteurs à neutrons rapides. Elles ont entre autres pour objectif l'étude du comportement du combustible, du réfrigérant et du confinement en cas d'accident, et portent principalement sur:

- la cinétique des défaillances du combustible et des gaines et leur propagation dans les boucles à sodium du réacteur CABRI: boucles SCARABEE (1 aiguille combustible vierge); SCARABEE IRRADIE (1 à 7 aiguilles irradiées); BETHSABEE (91 aiguilles organisées en 7 assemblages);
- les réactions sodium-combustible: expériences hors pile de Grenoble au Service des transferts thermiques prévoyant entre autres la possibilité de la fusion de 5 kg d' UO_2 , et expériences en pile dans CABRI;
- le comportement des produits de fission dans le sodium;
- l'ébullition du sodium: expériences hors pile de Grenoble (boucle mono-aiguille CFNa) et de Cadarache (boucle à 7 aiguilles bulle doseur), expériences en pile dans CABRI et RAPSODIE;
- les réactions sodium-eau dans les générateurs de vapeur;
- les incendies de sodium;
- les explosions et les réponses des structures aux sollicitations dynamiques.

En ce qui concerne les réacteurs graphite-gaz, les études encore en cours concernent les conditions d'incendie des gaines des éléments combustibles et le comportement de ces derniers après rupture de gaine. Elles sont conduites à la fois sur les plans théorique et expérimental (boucle hors pile pour l'incendie, essais en pile dans les boucles autonomes du réacteur PEGASE pour le comportement du combustible avec gaines percées, écriture et utilisation de codes de calcul).

Les combustibles des réacteurs à haute température diffèrent de ceux des réacteurs graphite-gaz par leur nature, mais les études sont de même type: comportement de l'ensemble et « bilan de transport » des produits de fission lors des accidents de refroidissement. En outre, des études particulières au cycle direct sont en cours.

Les problèmes de mécanique constituent aujourd'hui un domaine majeur des préoccupations en matière de sûreté parce qu'ils sont à l'origine de nombreux incidents liés à la défaillance de parties « non nucléaires » des centrales de puissance, ou à des erreurs d'exploitation. Historiquement, la construction des réacteurs a été marquée par des difficultés d'ordre surtout physique dans la phase initiale des réacteurs expérimentaux (neutronique, dynamique, thermique, contrôle et instrumentation, protection) et d'ordre métallurgique au stade des prototypes de puissance (comportement des matériaux sous irradiation, corrosion, fatigue). Parmi ces problèmes,

la corrosion reste un facteur important (corrosion des assemblages boulonnés dans le CO_2 , corrosions aqueuses diverses sous tension ou cyclages mécaniques des matériaux du cœur ou des structures internes).

Cependant, les défaillances les plus nombreuses concernent surtout les mécanismes, les pièces tournantes et les composants statiques, en particulier les assemblages soudés. Ces problèmes de technologie «classique» prennent une plus grande acuité dans le domaine nucléaire parce que les difficultés que présente l'exécution des réparations exigent que soit assuré d'emblée, dès la construction, un niveau de fiabilité très supérieur à celui que l'on accepte en technologie ordinaire. Cette exigence est particulièrement délicate pour les composants statiques (comme l'enveloppe du circuit primaire), dont la fiabilité est très difficile à évaluer car, si, de par leur nature, leur taux de défaillance est très bas, les conséquences de leur rupture pourraient être très importantes.

Il est communément admis dans le monde que la possibilité d'une rupture des tuyauteries reliées aux grands corps de pression en acier doit être prise en compte, mais on considère en général comme incroyable la rupture de ces derniers en raison des considérations suivantes: pressions dynamiques plus élevées dans les tuyauteries (vitesse de circulation des fluides supérieure); possibilité de vibrations en flexion couplées avec des phénomènes hydrodynamiques ou aérodynamiques dans les tuyauteries de grande longueur; fatigue par vibrations plus à craindre dans les tuyauteries (fréquence élevée d'où temps de détection d'une fissure avant rupture faible); possibilité de fissures circonférentielles donnant lieu moins facilement à une fuite avant rupture; rupture des tuyauteries constatée plus fréquemment que celle des cuves.

Cependant les grandes cuves en acier sont, par rapport aux tuyauteries, plus difficiles à construire, à contrôler en fabrication et en service, à instrumenter ou à réparer. Les nombreuses soudures aux raccords sont délicates et le matériau se fragilise en service. L'hypothèse de non-rupture doit donc être assortie de précautions extrêmes à la construction (qualité) et en exploitation (surveillance, essais et inspections périodiques, etc.) que l'on doit s'efforcer de systématiser, par exemple en élargissant les codes du type ASME (section III). L'expérience courante permet d'attribuer aux grands corps de pression une probabilité moyenne de rupture de l'ordre de 10^{-5} alors que l'on pourrait désirer 10^{-8} pour une cuve de réacteur, mais il n'est pas aisé, aujourd'hui, de démontrer que le surcroît de précautions couvre les 10^{-3} additionnels. En tout état de cause, les risques classiques normalement encourus par les populations sont d'un niveau très supérieur.

L'analyse des caissons en béton précontraint est plus rationnelle. Les fonctions demandées (étanchéité, résistance, isolement thermique) sont remplies par des organes relativement indépendants et qui peuvent être surveillés assez facilement. La résistance aux effets d'entaille, si préjudiciables aux structures homogènes, est grande (armatures multiples). Une rupture sans avertissement ne serait concevable qu'en cas de corrosion systématique des structures intéressant une surface de coupe bien déterminée (sinon il y aurait des variations dimensionnelles détectables). Une fuite primaire causant cette corrosion ne peut échapper à la détection; le caisson, d'autre part, est protégé des intempéries. Cette absence de facteur systématique d'agression possible et les expériences réalisées sur maquette justifient la pleine confiance accordée aux caissons en béton précontraint construits

suivant les règles de l'art. Sur un plan plus général d'autre part, la technique du béton précontraint est très riche en améliorations virtuelles pour les divers types de réacteur et justifie des programmes de développement technologique que les organismes de sûreté peuvent utilement promouvoir.

Dans le domaine de la sûreté radiologique, parmi les problèmes qui se posent dans la perspective de l'installation d'un grand nombre de réacteurs de puissance, les efforts les plus importants ont été orientés dans trois directions principales:

- le transfert de radionucléides à travers le sodium vers l'argon (dans le cas particulier des réacteurs surgénérateurs refroidis par le sodium),
- la dispersion des produits radioactifs dans les installations ou parties d'installation,
- l'étude de la diffusion atmosphérique à l'aide de traceurs.

a) Pour la filière à neutrons rapides, un effort tout particulier a été fait pour préciser le comportement des produits radioactifs dans le circuit primaire, notamment leur passage du sodium vers l'argon. En effet, on est amené à accepter qu'un réacteur puisse fonctionner avec un certain nombre d'aiguilles défectueuses, présentant des défauts d'étanchéité. Dans ces conditions, les produits de fission, particulièrement les gaz nobles (et les produits gazeux ou liquides à la température du combustible) contaminent le sodium puis diffusent dans l'argon. Comme il est impossible d'éviter toute fuite d'argon, il est nécessaire d'évaluer le niveau de contamination que peut atteindre ce gaz afin de prévoir les conséquences radiologiques de ces fuites. Des mesures ont été réalisées dans une boucle hors pile (expérience PIRANA), qui ont montré que 80% des gaz sont retenus par le sodium. Mais il faut vérifier si cette rétention reste aussi importante dans le sodium en mouvement.

Ces expériences permettent aussi d'évaluer les conséquences radiologiques d'un accident dans le cœur du réacteur, accident conduisant à la libération brutale d'une quantité importante de gaz rares. Simultanément les informations obtenues au cours de ces expériences permettent de mieux définir les conditions de réalisation du système de détection de ruptures de gaine.

b) Dans l'évaluation de la sûreté radiologique d'une installation nucléaire, il est important de connaître le comportement des produits radioactifs dans les différents circuits qui peuvent être contaminés aussi bien en fonctionnement normal (circuit primaire) qu'en cas d'accident (circuit secondaire, circuit de ventilation, enceinte, locaux divers). Ce genre de problème est indépendant de la filière mais chaque type de réacteur est un cas particulier. Dans ce domaine, des études théoriques visent à relier l'activité que l'on peut mesurer en un point accessible à la mesure à l'activité émise au niveau du combustible en tenant compte des caractéristiques propres du circuit (diamètre des canalisations, existence de coudes, vitesse de circulation), du fluide (viscosité, densité, température) et des produits concernés (état physique, densité, diamètre des particules). En ce qui concerne le transport par les gaz dans les gaines de ventilation, des expériences sont prévues pour vérifier les modèles théoriques établis et évaluer les quantités de produits radioactifs qui seront finalement retenues par les filtres.

c) Les accidents pris en considération dans les rapports de sûreté se traduisent finalement par la libération dans l'atmosphère d'une certaine quantité de produits radioactifs. L'évaluation des conséquences pour la

population comprend l'estimation des concentrations que l'on peut atteindre compte tenu des activités rejetées.

A cette fin on utilise souvent les formules théoriques classiques établies, soit par Sutton, soit par Pasquill. Il nous a paru préférable d'essayer d'atteindre les coefficients de diffusion par voie expérimentale. Des abaques ont été établis qui permettent de faire une première évaluation des concentrations que l'on pourrait atteindre aux alentours du point d'émission. Après la construction on peut procéder à des vérifications qui permettent de tenir compte de la turbulence créée par l'installation elle-même et les constructions avoisinantes, les émissions d'un traceur (uranine) ayant lieu dans des conditions météorologiques variées.

De plus, le relief autour du site et la nature de la couverture végétale éventuelle sont « intégrés » dans ces expériences in situ alors qu'il est très difficile de faire entrer ces variables dans les formulations théoriques.

CONCLUSIONS

Coûteuses mais indispensables pour pouvoir le moment venu utiliser sans restriction une méthode d'analyse quantitative, les études de sûreté constituent un domaine où la collaboration internationale devrait pouvoir jouer à plein. Les travaux que le CEA a récemment entrepris en commun avec des organismes étrangers dans plusieurs domaines se rapportant à la sûreté des réacteurs sont prometteurs dans cet ordre d'idée.

CANADA JUDGES POWER REACTOR SAFETY ON COMPONENT QUALITY AND RELIABLE SYSTEM PERFORMANCE

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Abstract—Résumé—Аннотация—Resumen

CANADA JUDGES POWER REACTOR SAFETY ON COMPONENT QUALITY AND RELIABLE SYSTEM PERFORMANCE.

For siting and safety of nuclear plants, Canadian authorities continue to base their judgements on upper limits set for (i) radiation exposures in routine and accident situations; (ii) plant failure rates and non-availability for safety systems. These factors are given special emphasis in Canada but, as elsewhere, approval also requires that the barriers to fission-product release be protected by design margins and safety devices. Component quality is specified in designing a system for a particular level of availability. It is embodied in procurement specifications and checked before installation. However the attainment of the level of availability must be demonstrated by periodic in-service tests. Severe accidents are postulated, including the simultaneous failure of a process system and a safety system (a 'double accident'). Under such conditions, safety devices must limit fuel failures and the shut-down capability keep radiation exposures within the set limits. In-reactor and out-reactor work has been extended to check the thermal and hydraulic performance of fuel and cooling systems, especially for transient conditions. Failure mechanisms for pressure tubes have been examined experimentally with material representing the range of metallurgical conditions to be found during the reactor life. Quality and other controls ensure necessary margins against failure, and experiments have shown the improbability that tube failures could propagate through the core. A description of the environment, including population distribution, plays its usual role in studies leading to site approval, but Canada is now seeking to define a standard site in terms of a statistical diffusion parameter.

LA SECURITE DES CENTRALES NUCLEAIRES CANADIENNES EST FONDEE SUR LA QUALITE DES COMPOSANTS ET SUR LA FIABILITE DES SYSTEMES.

Les autorités canadiennes continuent d'évaluer l'implantation et la sécurité des centrales nucléaires en basant leur jugement sur les limites supérieures des paramètres suivants: radioexpositions dans les situations normales et accidentelles; taux des pannes dans la centrale et des indisponibilités des systèmes de sécurité. Le Canada attache beaucoup d'importance à ces paramètres mais, comme ailleurs, il faut également, pour obtenir les permis, que les barrières contre l'échappement des produits de fission soient protégées par des marges prévues dans les plans et par des dispositifs de sécurité. La qualité des composants est spécifiée lorsqu'on fait les plans d'un système devant avoir un niveau de disponibilité donné. Cette qualité est incorporée dans les spécifications d'achat et elle est vérifiée avant installation. De plus, des essais périodiques, en cours de fonctionnement, doivent permettre de démontrer que le niveau de disponibilité voulu a été atteint. On suppose de graves accidents, y compris les pannes simultanées d'un système de fonctionnement et d'un système de sécurité (un «double» accident). Dans de telles circonstances, il faut que les dispositifs de sécurité limitent les ruptures du combustible et que la réponse des systèmes d'arrêt d'urgence restreigne, dans les limites prévues, les radioexpositions. D'importants travaux ont été effectués en et hors pile pour étudier le comportement thermique et hydraulique des combustibles et des caloporteurs, surtout dans les conditions transitoires. Les mécanismes de la rupture des tubes de force ont été étudiés au moyen d'expériences faites avec des matériaux simulant la gamme des conditions métallurgiques pouvant se produire durant la vie prévue d'un réacteur. Des contrôles de qualité et d'autres vérifications assurent des marges de sécurité et des expériences ont montré qu'il est peu probable que la défaillance d'un tube puisse se propager au

travers du cœur. Pour le choix des sites on a généralement recours à une description de l'environnement tenant compte de la répartition de la population. Cependant, le Canada cherche maintenant à établir un type standard de site dépendant d'un paramètre de diffusion statistique.

В ОСНОВЕ РЕШЕНИЙ ПО БЕЗОПАСНОСТИ ЭНЕРГЕТИЧЕСКОГО РЕАКТОРА В КАНАДЕ ЛЕЖАТ КАЧЕСТВО УЗЛОВ И НАДЕЖНАЯ РАБОТА СИСТЕМ.

При выборе площадок для строительства ядерных установок и при оценке их безопасности канадские учреждения продолжают при вынесении решений по безопасности основываться на верхних предельных значениях, установленных для: (а) радиационного облучения при повседневных и аварийных ситуациях; (б) частоты выхода оборудования из строя и его неготовности к работе в системах аварийной защиты. В Канаде этим факторам придают особое значение, но, как и в других странах, для получения разрешения на строительство также требуется, чтобы средства, предотвращающие выброс продуктов деления, имели бы защиту, обеспечиваемую как проектными расчетами, так и устройствами аварийной защиты. При проектировании системы с учетом определенного коэффициента использования указывается качество узлов оборудования. Оно указывается в спецификациях на поставку оборудования и проверяется перед монтажом. Однако достижение заданного коэффициента использования необходимо демонстрировать путем проведения периодических испытаний работающего оборудования. Определены наиболее серьезные аварии, включая одновременный отказ технологической системы и системы аварийной защиты (двойная авария). В этих условиях устройства аварийной защиты должны ограничивать величину неисправности в топливных элементах, а также обеспечивать срабатывание системы остановки реактора в случае превышения заданных пределов для доз радиационного излучения. Работы, проводившиеся как в реакторе, так и вне реактора, были расширены с включением проверки тепловых и гидравлических характеристик топлива и систем теплоносителя, особенно в условиях переходных режимов. Механизмы появления повреждений в трубопроводах высокого давления были экспериментально исследованы на материале, представляющем возможность исследовать ряд металлургических условий, с которыми приходится встречаться в течение всего срока службы реактора. Контроль качества и другие формы контроля обеспечивают необходимый запас прочности при аварии. Эксперименты показали отсутствие вероятности того, что повреждения трубопроводов могут распространяться по активной зоне. Описание окружающей среды, включая рассеяние населения, имеет свое обычное значение при исследованиях, предшествующих получению разрешения на участок для строительства, но в настоящее время в Канаде предпринимаются усилия для определения стандартного участка для строительства реактора с точки зрения статистического параметра рассеяния.

CANADA ESTIMA LA SEGURIDAD DE LOS REACTORES DE PRODUCCION DE ENERGIA POR LA CALIDAD DE SUS COMPONENTES Y POR LA FIABILIDAD EN EL FUNCIONAMIENTO DE LOS SISTEMAS.

Para los emplazamientos y seguridad de las centrales nucleares las autoridades canadienses continúan fundando sus juicios en los límites máximos fijados para a) dosis de exposición a la radiación en situaciones normales y de accidente, y b) índice de frecuencia de averías de la central y de no disponibilidad de los sistemas de seguridad. En Canadá se da una importancia especial a estos factores aunque, como en otras partes, la autorización requiere también que las barreras contra la emisión de productos de fisión estén protegidas por márgenes de proyecto y dispositivos de seguridad. La calidad de componentes se especifica al proyectar un sistema para un nivel determinado de disponibilidad. Se incluye en las especificaciones de adquisición y se verifica antes de su instalación. Pese a todo hay que demostrar mediante ensayos periódicos durante el funcionamiento el grado de disponibilidad conseguido. Se postulan accidentes graves, incluso el fallo simultáneo de un sistema funcional y un sistema de seguridad (un «accidente doble»). En tales condiciones los dispositivos de seguridad tienen que limitar los fallos en el combustible y la capacidad de apagado del reactor tiene que mantener las dosis de exposición a la radiación dentro de límites prefijados. Los trabajos que afectan al interior y al exterior del reactor se han ampliado para comprobar las características térmicas e hidráulicas de funcionamiento de los sistemas del combustible y refrigeración, especialmente en régimen transitorio. Se han experimentado los mecanismos de fallo de los tubos de presión con material representativo de toda la gama de las condiciones metalúrgicas que puedan encontrarse durante la vida del reactor. Los controles tanto de calidad como de otro tipo garantizan los márgenes necesarios frente a posibles fallos y experimentalmente se ha comprobado la improbabilidad de que los fallos en los tubos se propaguen por el núcleo. La descripción del medio circundante, incluyendo la distribución de población, desempeña el papel normal en los estudios encaminados a autorizar un emplazamiento, aunque Canadá intenta definir un emplazamiento normalizado en función de parámetros estadísticos de difusión.

1. INTRODUCTION

Atomic energy in Canada is a federal responsibility under the Atomic Energy Control Act. The licensing of nuclear power plants is one of many functions that fall to the Atomic Energy Control Board (AECB) under this act [1]. The Board proper comprises five members, one of whom is designated as its president, and the regulations are administered through the president and his staff. Because the staff is relatively small, it does not carry out an extensive reevaluation of a design when considering an application for a licence as is sometimes done elsewhere, notably as appears to be the case in the United States. However, the staff is competent to assess on behalf of the Board the characteristics of nuclear power plants from their conception and throughout their life.

A Reactor Safety Advisory Committee (RSAC) making extensive use of the Board's staff, advises the Board proper on every application. An operating licence is granted after lengthy discussions between the applicant and the RSAC, during which detailed technical presentations of the safety analysis are made.

Before deciding to grant a licence, the Board must be satisfied that:

- the necessary competence for the design and procurement of a nuclear station was available to the applicant;
- adequate information has been submitted concerning the design and site;
- appropriate assurance has been given for the performance of the reactor;
- sufficient controls have been applied in the design, component manufacture and installation phases to ensure the desired performance; and
- the administrative, operating and maintenance disciplines appropriate to the use of the plant are documented and will be put into effect.

The RSAC and the Board staff obtain this satisfaction by discussion with the applicant and by examination of the necessary documents and facilities. No attempt is made to duplicate the applicant's detailed analysis. Self-governance by the applicant, which encourages co-operation with the authorities and induces self-discipline and responsibility, is an essential element of the Canadian licensing procedures. A permit must be obtained before construction of a nuclear power station can begin. Operating licences are renewed on a periodic basis dependent upon a continuous review of reactor operations.

2. REACTOR SAFETY CRITERIA

The AECB has not attempted to give design and siting requirements in detail; instead, it fulfils its function by means of a statement of general principles, leaving to the applicant the freedom to implement these principles in a way that will achieve the objectives embodied in the principles.

The main objective is stated in terms of tolerable levels for releases to the environment and for the radio-exposure of people, during both

normal routine operating conditions and certain other unusual reactor conditions. The control levels from which the environmental release levels may be derived are given in Appendix A to this paper. These levels of radio-exposure of people are in terms of both an individual and a population risk. The level for normal conditions is based on the International Commission for Radiological Protection (ICRP) recommendations. For accidental conditions the intent is to limit the biological effect to what is considered an acceptable level [2,3]. Other things being equal, the control of routine releases is predominantly a matter of economics, because any additional control can be had by an increase in hold-up capacity or hold-up time, by additional filtration or by other means, but accidental releases are most likely to arise because of some complex interaction between characteristics of the design, the procurement controls and the operating conditions. These contributing conditions must be balanced through the overall safety philosophy that is applied to a nuclear power station. This safety philosophy is bounded by certain general design principles, which are stated in terms of limits on failure rates of components and the reliability and unavailability of systems as well as in terms of a required ability to control the consequences of certain accidental sequences. Complete independence between the power regulating system and the reactor emergency shut-down systems has been made a necessary prerequisite for an acceptable design and this is intended to ensure that serious accidents will arise only from a combination of events that are themselves very unlikely.

The general principles indicate how and what unlikely events are to be considered in establishing the safety design for a power station. This paper will not debate the level at which the controls have been set: it will be sufficient to say that the level of safety expected from the controls can be achieved by state-of-the-art engineering of high quality (see section 5).

3. PERFORMANCE STANDARDS

The performance standards required by the AECB are related to the two recognizably distinct functions into which a nuclear plant can be divided. The one function comprises all the process systems, which includes the nuclear furnace, the heat transport system, the reactor power regulating system and the electrical generating system. The other function comprises the protective systems, which includes those parts intended to limit the radiation effects of failures occurring in the process systems.

The Board has stated a standard for the maximum acceptable failure rate for the process systems. The standard requires that dangerous process system faults should not occur at a frequency greater than 1/3 per year; a dangerous process fault is one that would require some protective action if the limits on radioactive releases into the environment are not to be exceeded. This standard applies to the sum of all dangerous faults and therefore should not be considered as a design criterion for specific very severe failures. Failures such as would be typically associated with serious loss-of-coolant accidents will not remotely approach this rate because the pressure-coolant envelope will be additionally controlled by the supplementary use of such piping and pressure vessel codes that the expected

interval between severe failures is likely to be in the order of 10 000 yr. Failure rates for the individual components and systems will in practice be graded by specific standards which relate to the severity of the fault, that is, to the level of protection the fault would demand.

The standard for process system reliability is intended to ensure a high quality of detail design and so reduce the demands which will be made on the protective systems. The process system fault rate could have been stated as a requirement that demands for protective action coming from process system faults should not exceed 1/3 per year. The availability targets for the protective system, given below, are considered to provide sufficient protection for this demand rate.

It is important to recognize that when a generating network has a large nuclear component, the overall standard of reliability likely to be required of the nuclear generating units will not be markedly different from that required of the systems important to safety. The generating and safety requirements should therefore be mutually reinforcing with regard to the attainment of the requisite quality.

The availability for the protective systems, which, as has been said, is related to the reliability of the process systems, is that each protective system shall be available for 99.7% of the time that such protection may be required. The system that comprises the containment concept is to be considered as a single protective system and an availability of 99.7% must be shown for that combination of systems necessary to ensure containment for a given accident situation.

The standard of effectiveness for the various protective systems is derived from the accident design criteria of the next section.

In addition to the general principles already outlined, certain other design principles also apply, the most important ones being that

- all protective systems shall be designed so that they may be readily tested at a frequency sufficient to demonstrate the required availability; and
- the automatic shut-down system(s) shall be independent of the reactor power-regulating system and of each other if there are more than one. The shut-down systems shall provide reliability equivalent to that expected from a two-out-of-three-channel coincident system.

4. ACCIDENT DESIGN CRITERIA

For the purposes of design and analysis, two general accident situations are described and design dose limits are stated for each.

4.1 Single failure

A process system fault, controlled by the proper operation of the protective systems. The exposure to individuals or the population at risk shall not exceed the annual limit for normal operating given in Appendix A.

4.2 Double failure

A process fault associated with failure of any one protective system. The limits for accidental exposure given in Appendix A shall not be exceeded. In this context:

- if there is an independent and complete duplication of a protective function, only one need be assumed to fail providing that it can be confidently predicted that an event invalidating one will not affect the other; and
- if the containment system or any part of it is assumed to be ineffective, the remaining protective devices shall be sufficient to control the worst process fault so that the accidental exposure limits will not be exceeded. The expression "Ineffective Containment" is not in the sense of there being large holes in the structure, but to avoid too absolute a reliance on containment, recognizing, e.g., the probability of ventilation dampers failing to close or of sealing devices deteriorating between inspections.

The annual individual exposure limit (see section 4.1), when associated with a ground level release, a child at 1 km, and Pasquil F weather, translates into an allowable release to the environment of about 2 Ci of ^{131}I .

The accidental individual exposure limit (see section 4.2), when associated with a ground level release, a child at 1 km, and Pasquil F weather, translates into an allowable release to the environment of about 200 Ci of ^{131}I .

5. PROCUREMENT QUALITY

State-of-the-art engineering of high quality (section 2) means that the necessary codes, standards and specifications for the design, manufacture, control of quality and assurance of quality will have been established by the applicant as appropriate to the use of the device and the risk to third parties as generally recognized by the authorities and by the industry itself. This reflects the attitude in law that if an abnormal risk is known to exist, action must be taken to avoid it.

Canada has adopted, to the extent appropriate and applicable to the Canadian nuclear power concept, the nuclear codes and standards of the USA, and this will continue as new U.S. codes and standards are developed. In particular, the practices for pressurized heat transport systems on Canadian reactors are based on the codes of the American Society of Mechanical Engineers (ASME) and the American National Standards Institute (ANSI). Not only are these codes representative of world-wide practices, but Canada has contributed to their development and they also form part of the legal basis for the regulation of the manufacture and use of pressure vessels in all Canadian provinces. However, it is recognized that the special nuclear codes issued by ASME are not for general application to all reactor types. They have evolved from the special requirements of U.S. water reactors using steel pressure vessels and they are also conditioned by the expected consequences of coolant-system failures in such reactors.

For Canadian reactors, a classification has been made of pressure components in the heat transport system outside the core based on the expected consequences of a failure of the individual components. The requirements of Section III (Nuclear Vessels) ASME Boiler and Pressure Vessel Code for the appropriate classification have then been applied. Special requirements were developed for the pressure-tube type core and these are described later.

So too, the new Section XI (In-Service Inspection of Nuclear Reactor Coolant Systems) of the ASME Code outlines procedures specific to U.S. Water reactors. The need in this important area of periodic reinspection of pressurized components is well known and well understood in Canada: the regulations of Canadian provinces require periodic inspection of pressure systems as may be decided by the Provincial Boiler Inspector. Differences in design, level of risk, and operating procedures give rise to requirements for periodic reinspection of pressure components on Canadian reactors that are different from those incorporated into the Section XI Code.

Generally, regulations that apply in Canada to the manufacture and use of pressurized components are made under provincial "Boiler and Pressure Vessel" acts, but when such components are to be used in nuclear power stations, it is the federal agency that must be satisfied. The Board requires that the provincial regulations be met but enters into discussions with the provincial agencies concerning the code or standard to be applied. In the due process of time, the criteria may well be incorporated into written Canadian codes by an appropriate body. The safety standard intended by such codes would be equivalent to that intended by the ASME Nuclear Vessel and Power Piping codes.

For general design, as it affects safety of the ancillary electrical, electronic and other type systems, specifications in use in the USA such as those originating with the Institution of Electrical and Electronic Engineers (IEEE) are generally used since in these areas Canadian requirements are similar to those of the U.S.

The control and the assurance of quality is achieved through the medium of material and manufacturing specifications determined by the applicant to the satisfaction of the authorities.

Specifications defining nondestructive testing methods and the qualification of nondestructive test operators have been established at a national level. A national code also exists for the approval of welding inspection agencies and is being developed to cover all engineering inspection agencies.

Personnel for nuclear power stations are examined and are granted licences by the AECB.

6. THE PRESSURE SYSTEM AND CORE DESIGN

The heat transport systems of the cores of Canadian heavy-water reactors are pressure-tube designs, using zirconium alloys in which the

pressure tube is thermally insulated from the moderator by a surrounding calandria tube. The choice of this arrangement was to avoid the uncertainties in the design and manufacture of pressure vessels of the size foreseen for the Canadian nuclear program. Since neutron economy was of the essence the efficiency of this reactor design concept depended heavily upon the wall thickness needed for the pressure tubes. The standards accepted by the authorities for the pressure-tube design incorporated margins of safety that were smaller than those required by the ASME codes current at the time. However, later issues of ASME codes for pressure vessels accepted similar reduced margins, on the basis of an improvement of control over the design and manufacturing functions similar to that applied to the pressure-tube design resulting in closer control of wall thickness as vessel sizes increase.

Certain operational conditions associated with the fuel cooling and geometry lead to a requirement for creep design to limit the increase in diameter of a pressure tube over its 30-year life to about 3%. This amount of creep is compatible with a design based on an operating stress of $1/3$ ultimate tensile at the working temperature [4,5,6] and is very much less than that for creep rupture.

Experiments have shown that a failure of a pressure tube would be almost entirely confined by the calandria tube surrounding the failure [5,6] and that a leak-before-break criterion will be met by the pressure-tube material throughout its in-reactor life [4]. Such cracks would be readily detected from the tritium in the ensuing heavy-water leak. However, the simple design and manufacture, which present no difficulties to nondestructive examination to ensure high and consistent quality, make the probability extremely remote that faults that could grow to an unstable size and so cause catastrophic failure would exist in the tubes as installed.

7. SHUT-DOWN CAPABILITY AND THE FUEL

The general reactor requirements establish a relation between shut-down capability and fuel performance. With large reactors it is not always practical to provide a shut-down capability fast enough to avoid any over-power being experienced by the fuel for the whole spectrum of process system failures. However, it has been possible to limit this over-power so that the safety margins provided in the fuel design to cover the transient demands of station operation are also sufficient for the accident cases. The Canadian requirements invoke the following specific accident situation, which among others must be covered by the design to limit fuel damage.

7.1

If, in the event of the worst rupture of the coolant system, the reactor is automatically shut down by the proper operation of a shut-down system and the other appropriate protective systems also work, it must be shown that:

- the design limits for accidental exposures will not be exceeded with the containment assumed to be very much below its designed effectiveness, i.e., a double failure (section 4.2); and
- the annual limit for routine exposures will not be exceeded with containment at its designed effectiveness, i.e., a single failure (section 4.1).

This means that, under what in many other countries are the conditions of "design basis accident" for water reactors, the release is expected to conform to normal operating effluent limits, which are very much lower than the generally accepted limits for releases under accident conditions [7].

If the reactor has only a single shut-down system, then in the event of the worst rupture of the coolant system, the shut-down system must also be assumed to fail and then it must be shown that the design limit for accidental exposures will not be exceeded with containment and other appropriate protective systems operating at their designed effectiveness.

In other countries, at least at the moment, the safety analysis for water reactors with containment does not seem to be required to account for a failure of the shut-down system combined with the worst loss-of-coolant event, nor to ask for duplicated shut-down systems to reinforce the position [7].

Papers at this conference [8,9] compare in-reactor experience with the expected performance of Canadian fuel and analyze defect experience. It is noted that the design procedures lead to fuel that can operate without failure, typically at 20–25% above the design power rating to terminal burn-up. Fuel that has been designed for normal operation at sheath temperatures of 300°C has been operated for extended periods without failing at temperatures above 500°C and at very much higher temperatures for periods that compare to the duration of over-power conditions in the accidental situation. Excursion into the dry-out region is not an immediate catastrophe and the long temperature–time constant of sintered uranium oxide fuel contributes a significant reduction to the rate of shut-down required. Operating experience shows that fuel of a uniform and high quality has been achieved [8]. In particular, the soundness of the Zircaloy sheathing is guaranteed by a sensitive and reliable ultrasonic inspection method developed for this purpose [10,11].

Rates of shut-down have been checked in full-scale and model tests. Moderator dump, shut-off rods, and poison injection are used variously on Canadian heavy-water reactors as shut-down methods.

8. RATE OF COOLANT BLOW-DOWN

Experimental studies have been made with models of a simplification of the reactor coolant system blown down through various sized orifices. The discharge from a complete rupture of a coolant header was found to give the fastest blow-down rate [12]. A theoretical solution to the problem assuming a constant friction factor and heat transfer coefficient yielded results that are in reasonable agreement with experiment. This

work has established an analytical—experimental correlation that includes friction and heat transfer and which has been used for the design of the shut-down devices and for the containment provisions.

Full-scale experiments on a hot-water loop in which the blow-down of a typical heavy-water reactor fuel channel was studied show that, in fact, blow-down is a slower process than predicated for the simpler system of the models.

9. CONTAINMENT

The three methods of pressure suppression that have been used illustrate the development of a containment system for the present large multi-unit nuclear stations. The first station used sequential pressure relief with water dousing; later, single-unit stations used water dousing alone; the newer multi-unit stations use a single separate vacuum reservoir for pressure suppression in addition to water dousing. The containment shells of the individual units are connected to this reservoir. In the event of a major loss-of-coolant accident, self-acting valves vent the affected reactor containment shell to the reservoir, which is maintained at a pressure (including the vapour pressure of the water in the tank) of approximately 0.07 bars. The reservoir is of such a size that it will accommodate the air and steam from a single reactor containment at atmospheric pressure (for the worst accident case). The connections between the reactor containments and the reservoir are large enough and the automatic valves open fast enough in response to a pressure rise in the containment building to limit the overpressure transient to under 20-s duration and to a peak pressure of 1.25 bars [13].

10. METEOROLOGY

The assumptions that are made for weather conditions have a very large effect in the calculation of allowable releases for routine and accident conditions. It is appropriate then to add some remarks on meteorology.

At the earlier nuclear-power reactor sites, a limited survey of the meteorological conditions was carried out. This mostly consisted of preparing a list of frequencies of wind direction from past records collected at the nearest station of the Meteorological Service of Canada.

A small micrometeorological survey was carried out at the site of the Douglas Point Generating Station and a number of short-term diffusion trials were conducted with fluorescent dye particles as tracer. Nevertheless, the basic philosophy is that derived working limits of discharges of airborne radioactive effluents from the station should be calculated with the assumptions that:

- for routine releases, the average diffusion rates are given by P. Bryant [14]; and
- for accidental releases, the pessimistic diffusion rates are as predicted by the use of the Pasquill diffusion equation and the parameters for class F weather.

In practice, the derived working limits for routine releases calculated using the first assumption are considerably larger (about 2 orders of magnitude) than the measured discharges from existing stations. This margin amply covers any differences that might exist in diffusion climatology between likely Canadian power generating sites and the average site described and used by Bryant.

In the case of accidental releases, the probable benefit from a modification of the assumption of Class F weather would not seem to justify the expense of the survey that would be required. It has been shown from a long-term study that large year-to-year variations in the frequency of occurrence of poor atmospheric diffusion rates occur at one site and it is pointed out that results of diffusion studies at specific sites may be quite misleading if they are confined to only one or two years [15]. The frequency of occurrence of the same poor diffusion conditions, a kilometre or more distant from the source, will not differ by more than a factor of 2 or 3 at different sites in the temperate latitudes, if long-term observations are used; however, the frequencies at the same site may vary by a factor of 100 or more from one year to another.

Local topographic features will affect diffusion rates only to distances of a few hundred metres. In Canada, an exclusion area having a radius of about 1 km has become standard for power reactor sites and is considered sufficient to provide for uncertainties covered by local variations in diffusion climatology.

11. CONCLUSION

Although the Canadian safety criteria allow credit for the capability of other protective systems that are intended to reduce the demands and reliance on the containment system, it is recognized that a cross-connected fault could destroy the independence of multiplicated systems. Redundancy coming from this independence is of course a predominant factor in the assessment of the reliability of protective systems and until such times as techniques are devised to show that, during operation, redundancy has not been destroyed by equipment malfunction or human error, containment is likely to remain a very important bulwark between the hazard and the population.

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APPENDIX A

RADIATION EXPOSURE LIMITS

1. NORMAL OPERATION

1.1 Individual member of public at exclusion boundary:

0.5 rem/yr whole body
5 rad/yr thyroid

1.2 Integrated exposure to population:

10^4 man-rem/yr
 10^4 man thyroid-rad/yr

(Integration is to be carried out for individual doses exceeding 5×10^{-3} rem/yr.)

2. ACCIDENT CONDITIONS (see text for application)

2.1 Individual member of public at exclusion area boundary:

25 rem whole body
250 rad thyroid

2.2 Integrated exposure to population:

10^6 man-rem
 10^6 man thyroid-rad

(Integration to be carried out for individual doses exceeding 0.5 rem)

(In applying 2. it shall be assumed that Pasquil type F weather prevails and the wind is directed towards the person at risk or the densest population.)

REACTOR SAFETY PHILOSOPHY AND EXPERIMENTAL VERIFICATION

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Abstract—Résumé—Аннотация—Resumen

REACTOR SAFETY PHILOSOPHY AND EXPERIMENTAL VERIFICATION.

All reactors are provided with equipment to prevent dangerous situations arising or to modify the potential consequences. For such measures to be effective it must be possible to predict the behaviour of the reactor under many conditions and to know whether the equipment supplied has the capability of performing adequately and reliably. On the one hand this requires the systematic use of experience relating to the performance of components and equipment, and on the other hand a co-ordinated research and development program to give advanced knowledge in heat transfer, fluid flow, nucleonics, fracture mechanics etc. Ideally the synthesis of safety by design and the analysis by experiment are finally confirmed by some measure of demonstration on the completed reactor. This paper provides some examples. The first example is concerned with the margins to dry-out in the Steam Generating Heavy Water Reactor in the event of flow failure, channel blockage or maldistribution of channel heating. Initial out-of-pile experiments led to confirmatory in-pile dry-out experiments on an actual fuel assembly. This confirmed the substantial margin of safety and showed that dry-out was a more stable and reversible process than might have been expected. The second example is concerned with the kinetic behaviour of the Prototype Fast Reactor (PFR) which is governed largely by the magnitudes of various reactivity coefficients such as those due to Doppler effects and sodium voiding. Experiments in the zero energy fast reactor ZEBRA have reduced the uncertainties in the coefficients and the results will finally be confirmed by experiments during the commissioning of PFR. The third example discusses the evaluation of the Doppler constant, the rate of reactivity addition, and the energy transfer from fuel to sodium as three important parameters affecting the design of containment.

PRINCIPES DE SURETE DES REACTEURS ET VERIFICATION EXPERIMENTALE

Tous les réacteurs comportent des dispositifs destinés à prévenir les situations dangereuses ou à en modifier les conséquences éventuelles. Pour que les mesures prises en vue de la sûreté soient efficaces, il faut que le comportement du réacteur puisse être prévu dans un grand nombre de situations et que l'on soit certain que le matériel fonctionnera convenablement et sans défaillance. Pour s'en assurer, on fait systématiquement appel à l'expérience du fonctionnement des composants et du matériel et on exécute des programmes coordonnés d'études et de réalisations permettant d'approfondir les connaissances sur le transfert de chaleur, l'écoulement des fluides, la physique nucléaire, la résistance des matériaux, etc. Dans le meilleur des cas, le concept de sûreté auquel on est parvenu par synthèse dans les études de plans et par analyse expérimentale est confirmé en dernier ressort par la démonstration sur un réacteur en fonctionnement. Le mémoire fournit quelques exemples de ce processus. Le premier exemple traite des marges de tarissement du réacteur à eau lourde producteur de vapeur, dans les cas de panne de circulation, de blocage de conduits, ou de mauvaise distribution de l'échauffement d'un canal. Les études préliminaires, effectuées hors pile, ont été complétées par des expériences de tarissement en pile sur un assemblage combustible réel. Celles-ci ont confirmé l'existence d'une marge substantielle de sûreté et montrent que le tarissement est un processus plus stable et réversible qu'il ne paraît. Le deuxième exemple concerne le comportement cinétique du réacteur rapide prototype (PFR), qui est régi pour une grande part par la valeur des divers coefficients de réactivité, comme les effets Doppler et le coefficient de sodium. Les expériences effectuées dans le réacteur rapide d'énergie nulle, ZEBRA, ont permis de réduire les marges d'erreur sur les coefficients; les résultats seront confirmés lors de la mise en service du PFR. Le troisième exemple est une discussion de l'évaluation de la constante de Doppler, de la vitesse de l'addition de réactivité et du transfert d'énergie du combustible au sodium, considérés comme trois paramètres importants qui influent sur l'étude de l'enveloppe de sécurité.

ПРИНЦИПЫ БЕЗОПАСНОСТИ РЕАКТОРОВ И ИХ ЭКСПЕРИМЕНТАЛЬНАЯ ПРОВЕРКА.

Все реакторы оснащены специальным оборудованием, предназначенным для предотвращения опасных аварийных ситуаций или для уменьшения возможных последствий аварий.

Чтобы эти меры были эффективными, необходимо уметь предсказать поведение реактора при различных условиях и знать, сможет ли данное оборудование надежно функционировать. С одной стороны, это требует систематической экспериментальной проверки работы соответствующих узлов и оборудования, а с другой — проведения координируемых научно-исследовательских и опытно-конструкторских работ для более детального изучения теплопередачи, динамики жидкости, ядерной техники, механики разрушения и т.д. В идеальном случае сочетание обоснованных методов проектирования и анализа экспериментальных данных обеспечивает надежную работу реактора. В данном докладе рассмотрено несколько примеров такого подхода к проблемам безопасности. Первый пример связан с запасом безопасности при процессе пересыхания в тяжеловодном реакторе SGHWR в случае сокращения расхода теплоносителя, блокировки каналов или неравномерного нагрева каналов. Результаты предварительных вне реакторных испытаний подтвердились затем при внутри-реакторных испытаниях по исследованию проблемы пересыхания на натурной тепловыделяющей сборке. Эти испытания подтвердили достаточно высокий уровень безопасности реактора и показали, что процесс пересыхания оказался более устойчивым и обратимым, чем ожидалось. Второй пример связан с кинетикой реактора-прототипа на быстрых нейтронах PFR, которая в основном определяется величиной различных коэффициентов реактивности, например коэффициентов, зависящих от эффекта Допплера и от образования пустот в натриевом теплоносителе. Эксперименты, проведенные в реакторе на быстрых нейтронах нулевой мощности ZEBRA, привели к уточнению этих коэффициентов. Результаты этих экспериментов должны быть затем проверены при пуске реактора PFR в эксплуатацию. В третьем примере рассматриваются три важных параметра, влияющих на конструкцию оболочки реактора, — оценка константы Допплера, скорость увеличения реактивности и перенос энергии с топлива на натрий.

LA SEGURIDAD EN EL REACTOR Y LA COMPROBACION EXPERIMENTAL.

Todos los reactores están dotados de equipo que impide las situaciones peligrosas y modifica sus posibles consecuencias. Para que estas medidas tengan efectividad debe ser posible predecir el comportamiento del reactor bajo diferentes condiciones y saber si el equipo suministrado tiene la capacidad para funcionar adecuadamente y con seguridad. Esto requiere, por un lado, el empleo sistemático de experiencias obtenidas relativas al rendimiento de los componentes y equipo y, por otro lado, un programa de investigación y desarrollo, coordinado, para proporcionar mayor información en la transferencia del calor, la nucleónica, el flujo de fluidos y la mecánica de fracturas, etc. Primordialmente, la síntesis de la seguridad en diseño y el análisis por experimento se confirman finalmente por demostraciones en el reactor completo. Este artículo proporciona algunos ejemplos. El primer ejemplo se refiere a los márgenes de desecación en el Reactor de Agua Pesada Generador de Vapor (SGHWR) en caso de falla del flujo, obstrucción en los conductos o mala distribución del calor del reactor. Los experimentos iniciales fuera del reactor condujeron a experimentos de confirmación a base de desecación en un conjunto real de combustible. Esto confirmó el margen substancial de seguridad y demostró que la desecación era un proceso más estable y más reversible de lo que se había esperado. El segundo ejemplo se refiere al comportamiento cinético del Reactor Prototipo Rápido (PFR) el cual está regulado principalmente por las cifras de los diferentes coeficientes de reactividad, tales como las que son debidas al efecto Doppler y a los vacíos del sodio. Los experimentos en el reactor rápido de potencia cero "ZEBRA" han reducido las incertidumbres en los coeficientes y los resultados serán finalmente confirmados por experimentos cuando entre en funcionamiento el PFR. El tercer ejemplo considera la valoración de la constante Doppler, el tipo de adición de reactividad y la transmisión de energía del combustible al sodio como tres parámetros importantes que influyen en el diseño de la contención.

INTRODUCTION

Some of the safety problems which loomed large in the 1950s no longer seem so important; they have been largely resolved or reduced to a manageable level by the research effort of the last twenty years. Some of the earlier uncertainties still remain, such as the behaviour of defects in primary pressure circuits or the transient effects which might follow a rapid depressurisation of a primary circuit. New problems also arise or old problems assume greater importance for new systems, such as the possibility of energetic reactions between hot fuel and coolant.

Research programmes which investigate these areas are concerned with very complex phenomena and it seems likely that they will continue for many years. As safety research is

expensive and requires a high standard of judgment in its direction and high quality of effort in execution it might be helpful to consider whether past experience gives any guide to the future choice of programmes and to their direction.

Though an exercise of this type is highly subjective, there is likely to be general agreement on the following points:

- (1) Safety problems are complex and may not have single all-embracing solutions,
- (2) Much valuable research has been of a fundamental type which progressively adds to knowledge and can be applied when required to urgent problems of the moment,
- (3) It appears in retrospect that some research has attempted to solve insoluble problems and that this effort might more usefully have been directed elsewhere.

Some further amplification of these points may be useful in setting the theme of this paper. This is now attempted by following the development of two apparently dissimilar areas of safety investigation which illustrate a common theme. These studies concern (a) cracks or defects in steel structures [1] and (b) thermal interaction between hot fuel and coolant [2].

The need for work is crystallised by an event or a series of events; in (a) through the brittle failure of ships and other similar events on land, and in (b) through the well-known SPERT event of 1962, together with explosions which have occurred in industry when quenching molten materials.

There follows a period of exploration in theory and experiment in the hope of finding a specific solution — that the cause can be established and the problem removed. In (a) the clue of brittleness influenced research programmes attempting to show that failure could not occur if the temperature of the structure remained above a suitably adjusted nil ductility temperature (NDT). In (b) there is interest in the factors influencing the efficiency of thermal exchange processes and some groups hope to obtain results from laboratory experiments to establish simple limit criteria analogous to the NDT concept.

Neither of these lines of safety research has led to a successful conclusion. The continued study of crack propagation shows that fast failure can occur at high temperatures above the NDT and interest has changed to the factors affecting critical crack length and the rate of crack growth. Again the analysis of a particular industrial accident in which molten steel fell into water and supporting evidence from experiments show that heat can be exchanged rapidly (in milliseconds) through the dispersion of molten materials in liquid coolants and that this process can lead to explosive vapour production.

In both case studies (a) and (b) the evidence shows that there is no general solution — fast failures and efficient thermal reactions can occur. It is also clear that a programme aimed at establishing conditions which could limit these events requires extremely advanced knowledge of the fundamental processes involved.

At an appropriate stage in the development of research programmes for (a) or (b) there is an interplay of imponderable questions such as:

- (i) Is it likely that further work on crack propagation will lead to an adequate description of the behaviour of any defect in any position in a pressure circuit?
- (ii) Is it likely that investigations of thermal reactions will explore sufficiently the significance of temperature, geometry and materials to enable limits to be applied to the many complex models of possible accidents?

Past experience suggests that a good understanding of fundamental phenomena is necessary to promote safety but that basic research may be prolonged with no certainty of a clear end point. If, then, current problems cannot at present be resolved by basic solutions, alternative means must be found to avoid or modify the problem or to limit its consequences. For instance, if the behaviour of defects in structures remains somewhat unpredictable, then there is a need for intensive effort to obtain signals from these defects. This route has been followed in recent years using sophisticated ultrasonics, self-generated stress waves and other techniques, so that in relation to problem (a) current research programmes are directed both towards the promotion of inspection techniques and towards the further study of fracture mechanics.

In case (b) if the upper limit of energy deposition is too high to be tolerated safely, then, as in (a), there is a search for signals from precursors to give warning that an event might occur if action is not taken. This route is being followed for the sodium-cooled fast reactor to identify the precursors of fuel coolant interaction, possibly by indications from coolant flow, sodium boiling, fission gas release, reactivity changes or other related phenomena.

It is sometimes the function of background research to establish boundaries to such variables as temperature, time, pressure etc., within which, detection or warning systems are required to work. In cases where the limits apply to near normal situations it may be possible to test the quality of the data and its interpretation by measurement made on the reactor itself, and the first example to illustrate this point is taken from tests on the Steam Generating Heavy Water Reactor (SGHWR).

The paper follows later with examples from the fast reactor illustrating the progressive interplay between theory, experiment and design.

SGHWR DRY-OUT EXPERIMENT

In any water-cooled reactor it is necessary to know when there is a significant loss of cooling due to steam blanketing of the fuel element surface. In a two phase system such as SGHWR or Boiling Water Reactor this condition is known as dry-out and is the point at which the surface water film is lost and all the liquid is entrained in the steam flow. The margin to dry-out must be estimated over the range of operating conditions. This margin is also important in assessing various fault conditions such as power failure to pumps and hence the time scale available for corrective action, the degree to which channel blockage might be tolerated, and the time to initiate emergency cooling.

At an early stage in the development of the SGHWR it was recognised that the prediction of dry-out in the 3.6 m long 36 rod bundles was going to involve considerable extrapolation from the existing test data. Moreover, there was little information concerning the influence of a cluster geometry, grids and power distribution on dry-out power. Accordingly a programme of work was mounted beginning with small single pin high pressure water rigs and leading up to full-scale experiments with electrically heated fuel assemblies in the 9 MW rig [3]. Tests with axial cosine heat fluxes, different degrees of pin peaking, tilted heat fluxes across pins, and tilted heat fluxes across the whole cluster were included in the programme. Some of the work was carried out in Freon rigs where the lower temperatures, pressures and powers made testing easier.

As it is impossible to simulate precisely all possible combinations of power distribution, flow, etc., likely to arise throughout the life of the reactor, a theoretical model which correlated the results and could be used to predict situations with different combinations of the variables was required. The HAMBO code was developed for this purpose [4]. This involves a two phase hydraulics model representing coolant flow in the array of coupled subchannels corresponding to the spaces between the fuel rods. Coefficients

representing mixing between subchannels were fixed empirically. Special dry-out correlations based on conditions in the subchannel where dry-out occurred were developed to fit a wide range of internationally published high pressure water data as well as the specific 9 MW rig SGHWR results.

However, the rig could never simulate all the reactor variables and it was decided to carry out full scale tests in the reactor, and in May/June 1970 a series of dry-out and post dry-out tests of a single full-size single-enrichment fuel assembly were carried out in a special loop within the Winfrith reactor itself. In the majority of cases dry-out was achieved by reducing channel flow while reactor power and channel inlet subcooling remained constant. Fuel cladding temperature excursions caused by dry-out were measured by means of thirty thermocouples attached to selected fuel pins at four axial levels in the top half of the cluster. Laboratory tests on single electrically heated tubes had previously shown that, when compared with the normal internally mounted resistance thermometers used in the 9 MW rig, these externally mounted thermocouples did not perturb the results.

More than 120 dry-out determinations were made for a range of channel powers between 4 and 6 MW and inlet sub coolings between 63 kJ/kg and 140 kJ/kg. The majority of tests were carried out with inlet pressure 6.2 MN/m² nominal; a limited number were carried out at 5.7 MN/m² and 5.2 MN/m².

Included in the above were several runs in stable post-dry-out conditions for periods up to 2 minutes at powers ranging between 3.9 and 5.2 MW. During the most extended run the flow was decreased in small steps to a value 26% lower than the initial dry-out value. Rapid flow reductions were also made into stable post-dry-out conditions and similar maximum temperatures were observed compared with those obtained at the end of a series of step reductions in flow. The results showed that dry-out occurs typically at about 35%, and that stable post-dry-out conditions were obtained at flows as low as 25%, of nominal channel flow rate. This would require a 95% area blockage at inlet or some lesser proportional blockage higher up the channel.

In all post-dry-out runs the cladding surface thermocouples did not record temperatures in excess of 600°C. Fig. 1 shows a typical post-dry-out run.

In all cases the tests were easily controlled and retreat from post-dry-out conditions was promptly achieved within 2 or 3 seconds when loop flow was increased. There was no evidence of the spread of the dry-out region when the flow was held constant; indeed the initial dry-out temperature excursions were self-quenching on most occasions.

Fig. 2 shows good correlation between the experimental results and the data from the 9 MW rig adjusted by the HAMBO correction and appropriate to the particular in-reactor conditions.

These tests in the reactor have confirmed the very considerable accuracy with which dry-out can be predicted over a range of conditions and have given increased confidence by showing how stable and reversible the dry-out phenomenon is with changes in flow or heat rating. The understanding thus obtained will permit SGHWRs to be operated with margins which are real factors of safety and not factors of ignorance.

REACTIVITY COEFFICIENTS IN THE PROTOTYPE FAST REACTOR

The second example considers the direct reactivity changes produced by some rearrangement of materials in the sub-assembly of a sodium-cooled fast reactor. These changes are compounded mainly of effects associated with modifying four aspects of the neutron economy: total fission rate, absorption rate, leakage, and energy spectrum. It is not

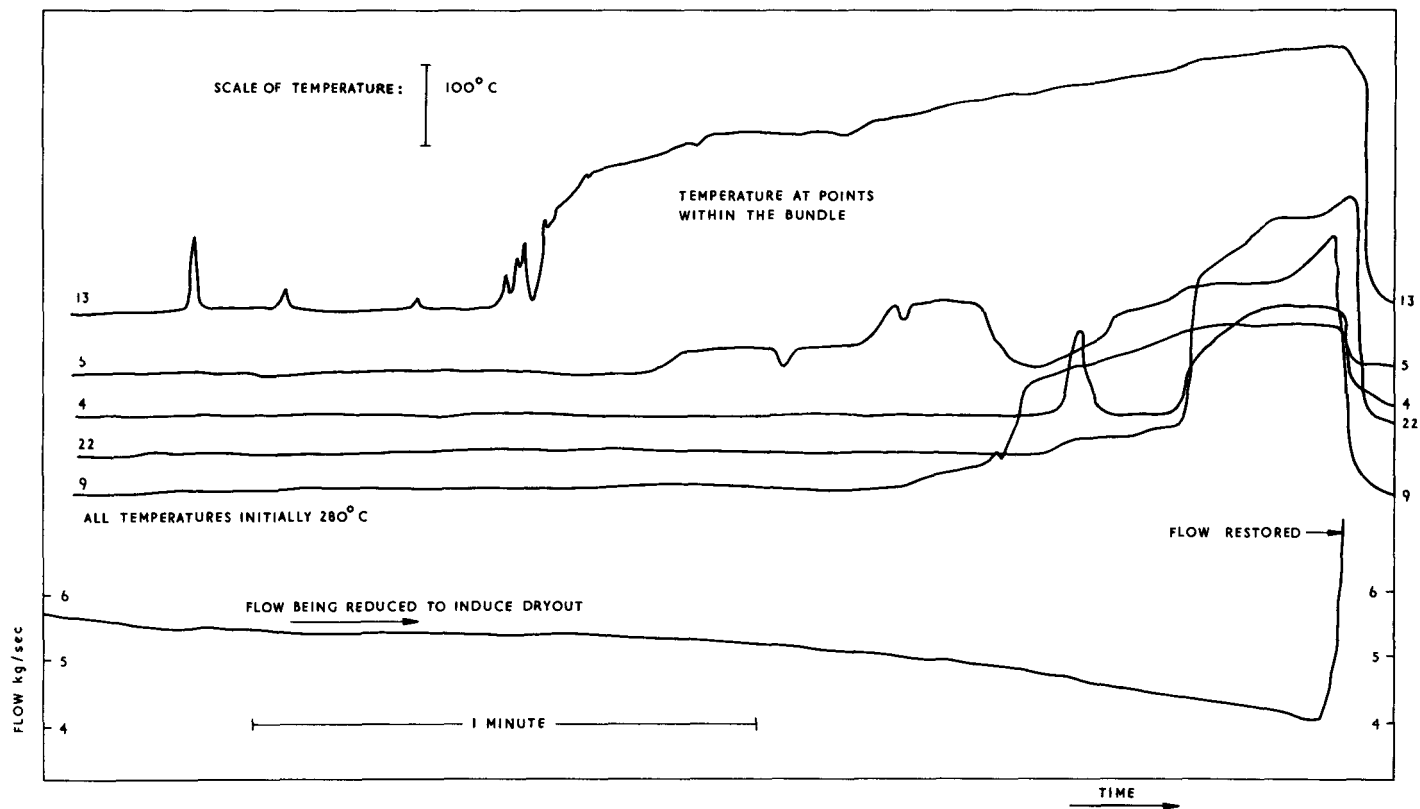


FIG. 1. Typical dry-out run in a single SGHWR channel.

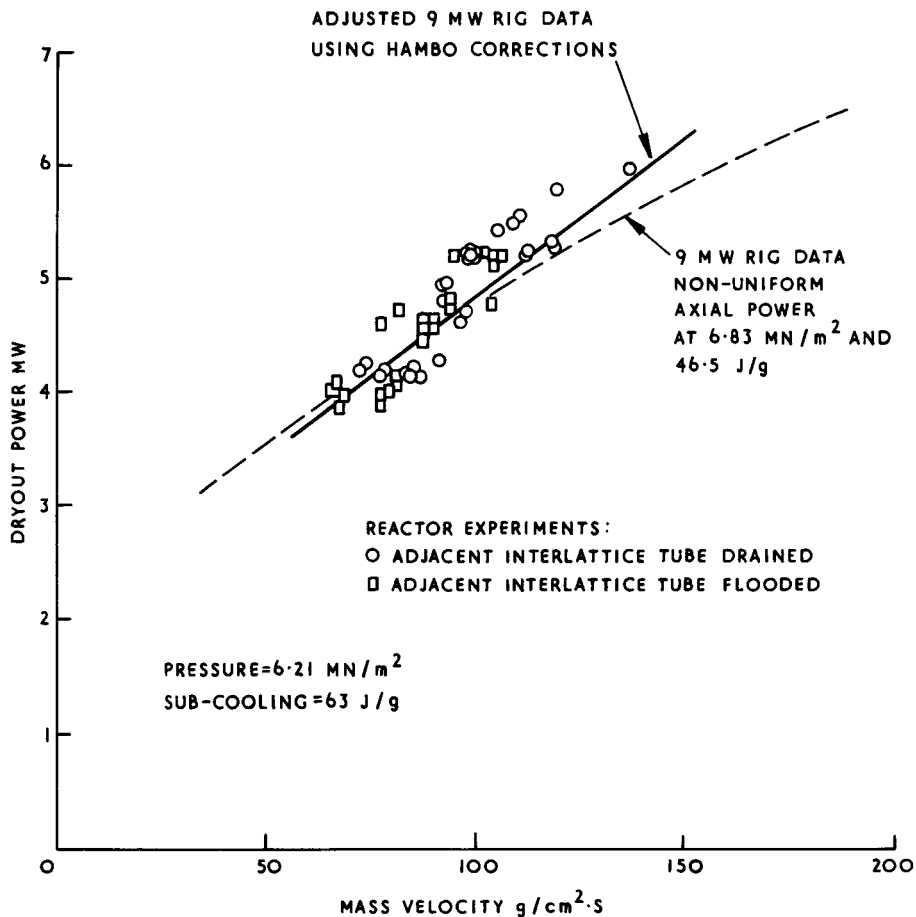


FIG. 2. Comparison of reactor dry-out experiments and out-of-pile data.

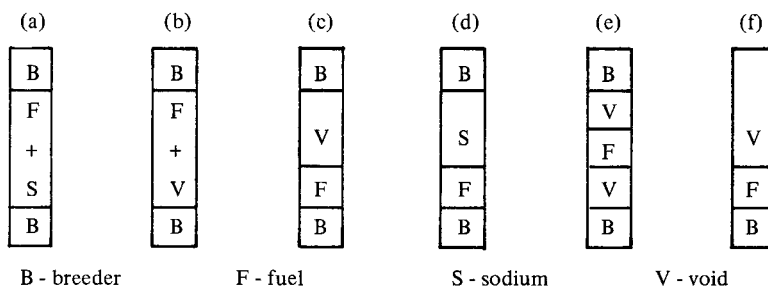
unusual for each of these four processes in isolation to produce effects which are an order of magnitude greater than their combined result, so that calculated reactivity changes are extremely sensitive to uncertainties in any of the four contributory processes.

In order to validate theoretical methods for calculating such changes, a series of measurements was performed in a ZEBRA core [5]. Whilst it was not possible to copy the PFR precisely, the core geometry and neutron spectrum were sufficiently well reproduced to ensure that a theory adequate for this ZEBRA core might be applied confidently to the PFR.

TABLE I
REACTIVITY CHANGES PRODUCED BY VARIOUS FAULT
CONFIGURATIONS IN PFR SUB-ASSEMBLIES

Configuration	Reactivity Change (%)	
	Experimental	Theoretical
(b) Loss of sodium	- 0.004 ± 0.001	+ 0.007
(c) Loss of sodium, fuel collapse to core bottom	- 0.020 ± 0.003	- 0.026
(d) Fuel collapse to core bottom, sodium re-entry	+ 0.032 ± 0.002	+ 0.028
(e) Fuel compacted at mid-core, sodium lost	+ 0.087 ± 0.018	+ 0.117
(f) Fuel at core bottom, sodium and upper breeder lost	- 0.274 ± 0.013	- 0.258

The six configurations illustrated below were built into the centre of the ZEBRA core, and the reactivity changes were measured relative to the normal sub-assembly. The measurements were calibrated by adding known quantities of fissile material.



- (a) Normal sub-assembly
- (b) Loss of sodium
- (c) Loss of sodium, collapse of fuel at core bottom
- (d) Re-entry of sodium above collapsed fuel
- (e) Fuel compacted at mid-core, sodium lost
- (f) Fuel compacted at core bottom, sodium and upper breeder lost.

The transport theory code, TURTLE, was applied to each in an S4 approximation [6]. Spectra and cell-averaged data were determined for each of the 33 zone compositions in the theoretical model in six groups condensed progressively from the 2000 group of the UK Fast Reactor Cross Section Set known as FLG4 [7]. The geometry was cylindrical with 17 radial and 44 axial mesh points. Theoretical and experimental results are compared in Table I.

The comparison of results is consistent with a standard deviation of $\pm 0.015\%$ in the calculation of reactivity. When the same theoretical techniques are applied to a similar rearrangement of fuel in a PFR sub-assembly a reactivity gain of 0.24% (or $\$0.80$) is predicted, whereas an automatic shutdown system can protect the reactor against reactivity changes of up to 0.33% ($\$1.10$). The precision achieved in calculating reactivity changes ($\pm 0.015\%$ standard deviation) in relation to a margin of 0.09% in this instance gives assurance that reactivity effects due to many rearrangements of fuel will not cause damage to the remainder of the core. This conclusion is not intended to be applied generally, nor is it presented here to support a particular design; in fact if the same sub-assembly were used in a larger civil fast reactor, as is proposed, the margins appear far greater. The story presented shows the relevance of information derived from research into notional accidents, and supports in this instance the conclusion that there are greater uncertainties in the accident model than in the data used in their analysis.

DOPPLER EFFECT IN FAST REACTORS

Turning now to the third example, the Doppler effect has long been recognised as a key factor in the safety of fast reactors. Between 1950 and 1960 it was important to show that the effect was negative and in 1955 results from the research reactor ZEUS, although inaccurate, indicated a small negative coefficient for the metal-fuelled Dounreay Fast Reactor. Subsequent interest has moved to larger ceramic fuelled reactors with more neutrons in the 1-10 keV range and in these reactors the Doppler effect is important in two main areas. The first relates to the economic design of a core and control system for which purpose a knowledge of the coefficient to within $\pm 15\%$ is desirable. The second relates to the action of the Doppler effect in a nuclear excursion in which a negative coefficient will limit the mean rate of energy deposition in the fuel and extend the timescale of the excursion into the millisecond range. This allows modest central pressures to disperse the core and reduce the overall consequences. In this accident the Doppler effect is being applied over a range of temperatures up to the boiling point of the fuel and in situations of partial or complete sodium voiding.

The main Doppler coefficient comes from a reduction in self shielding in the broadened capture resonances of ^{238}U , but contributions from other materials cannot be ignored. So far as ^{239}Pu is concerned it appears that the negative effect from capture almost completely balances the positive effect from the fission resonances, whereas ^{240}Pu and the canning material will contribute a small negative effect. These effects can all be individually calculated and the experiments that have been performed have been designed as checks on significant parts of the calculation. For example, a special experiment was mounted in ZEBRA in 1965 in which a large central zone of the reactor containing oxide fuel was heated to 550°C. [8]. Initial disagreements were resolved by direct measurement of the spectrum, and calculations of the Doppler constant then agreed with experimental results to within 15%. Additional confirmation of experiment and theory was sought through further experiments in the UK and elsewhere. These have included the activation of foils in a fast reactor spectrum measuring the effect of temperature on the rate of neutron capture and perturbation experiments in zero power assemblies comparing the reactivity worth of small samples at different temperatures [9].

Further support to the theoretical estimates is being obtained from the analysis of the behaviour of special experimental reactors such as VIPER and SEFOR [10]. VIPER is a small pulsed reactor fuelled with 37.5% enriched metallic uranium rods, and is made prompt critical by the rapid insertion of a fuel rod and shut down by its inherent negative temperature coefficient, a large part of which is due to the ^{238}U Doppler coefficient. Analysis of the results from this reactor have enabled a comparison to be made of the calculated and measured Doppler coefficients and once again reasonable agreement has been obtained;

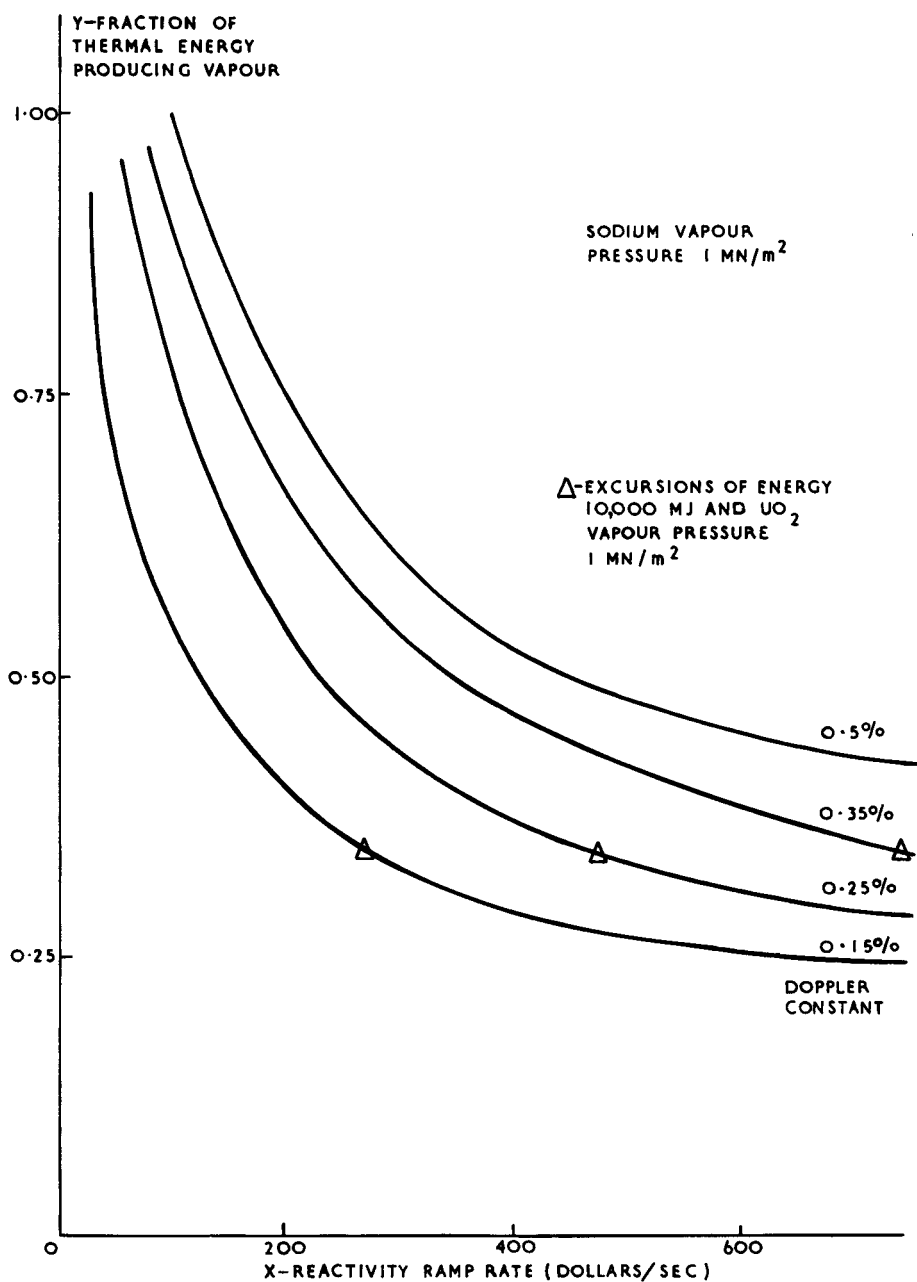


FIG. 3. Curves of X against Y for four different values of the Doppler constant.

calculation underestimating the measured value by about 15%. Measurements made during the operation of a large ceramic fuelled prototype reactor such as PFR will provide, for reactors of that type, sufficient information on the Doppler effect under normal operation.

The Doppler constant is only one of the factors affecting the size and consequence of a nuclear excursion. For the idealised models presently conceived, the rate of reactivity addition, together with the Doppler constant, determines the total energy input to fuel. The energy input to fuel, in turn, determines the pressure which the expanded UO_2 vapour may exert in the system in which it is confined (this may be a tank and roof structure); the extent to which this energy is passed from fuel to sodium determines the additional pressure which sodium vapour would exert in the same confined space. Whilst it may be an over-simplification to regard these as steady pressures, it is useful in this example to regard them as such, although in a more detailed analysis, transient effects are being studied to see if they are important.

The mixing of hot fuel with coolant is unlikely to produce the maximum pressure which is theoretically possible; Some vapour might condense in the surrounding body of cooled sodium and the mixing process may not bring together the optimum ratio of fuel and coolant. Exploratory experiments with a number of materials have given varying results and in many cases the ratio of pressure achieved to the maximum possible has been less than 10% and only a few results have exceeded this figure.

Three factors have now been mentioned which affect the excursion (there are others beyond the scope of the present paper). These three factors, the thermal energy conversion (Y), the rate of reactivity addition (X), and the Doppler constant, will define an energy input to the system and a vapour pressure within the confinement space. This pressure and any transients associated with it will be a determining factor in a design, and to illustrate the relative significance of the variables, a typical study was carried out and illustrated in Fig. 3 for a reactor having a mass of fuel of 16t and an available blanket gas space of $2,000 \text{ m}^3$.

The curves present for four different values of the Doppler constant the related values of X and Y which would yield a sodium vapour pressure of 1 MN/m^2 for the conditions of this study. It is seen that:

- (a) It is worthwhile ensuring that the Doppler constant is not below 0.2%. Higher values in the range 0.3% to 0.5% produce only a limited extension to the range of accidents which may be tolerated at the unlikely end of the accident spectrum.
- (b) High values of Y would severely limit the range of tolerable accidents determined by X, but if Y does not exceed 20% to 30% (and it may be very difficult to achieve higher efficiencies) then the pressure of sodium vapour is comparable with that which could be produced in the blanket gas by the expansion of the partially vapourised fuel due to the same ramp rate and Doppler value. The symbols Δ on Fig. 3 show when this condition applies.

If some other design provided a much smaller volume for the cover gas or could only tolerate a much lower pressure then:

- (a) There would be a greater interest in establishing confidence in a higher Doppler constant, although this alone would not remove very severe limitations on the permitted range of X and Y.
- (b) There would be an incentive to show that the 'likely' accidents were in the lower range of reactivity ramp rate - less than $\$100$ per second, and an interest in establishing credence in low thermal efficiencies (Y) of less than 10%.

This situation exemplifies a typical stage of development leading to questions of the type proposed earlier in this paper:

- (a) Is it likely that continued theoretical studies of reactor excursions will lead to a conclusion that high ramp rates of say greater than \$100 per second are unlikely and may be termed 'incredible'?
- (b) Is it likely that continued experimental studies into fast thermal exchange between fuel and coolant will in a comparable manner show that efficiencies of more than 10% are unlikely?
- (c) Should the designer at some cost provide greater tolerance and would such designs have an equal or better capability of dealing with other aspects of the excursion or with other accidents?
- (d) It is also fair to ask whether the design should take into account the possibility of an excursion or whether this event should be made 'incredible' by other appropriate design solutions.

CONCLUSIONS

Throughout this paper the authors have tried to illustrate the interplay between safety philosophy, and research and development programmes which both affect and are, in turn, influenced by current designs. There is a pattern of broadly based fundamental research into physical phenomena which will continue over many years. As reactor systems develop there is a recurring need to seek specific solutions to problems of the moment. The short-term programme which then emerges is strongly influenced by current understanding of the fundamental phenomena. Thus the long-term and short-term aspects of a research programme have compatible aims which should lead to successive improvements in reactor design and increased confidence in their safety.

All of those engaged in safety work recognise its complexity and will agree that safety research requires a high standard of judgment in its direction and a high quality of effort in execution. It seems therefore appropriate in this international forum to make two final points:

- (a) The diversity of research made by a number of countries into similar problems, and the varied interpretation of this work, can act in a positive and complementary manner, if well co-ordinated.
- (b) Safety is the concern of all countries and for the continued successful application of nuclear power it is possibly even more important that collaboration should take place in this field than in the commercial field – a theme discussed by Peirson in another paper [11].

ACKNOWLEDGMENTS

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TECHNICAL SAFETY ISSUES FOR LARGE NUCLEAR POWER PLANTS

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Abstract—Résumé—Аннотация—Resumen

TECHNICAL SAFETY ISSUES FOR LARGE NUCLEAR POWER PLANTS.

Although the basic requirements for and means of achieving reactor safety remain unchanged, much has been learned from research and development projects and from analysis and experience in the design, construction, and operation of power reactors. Safety reviews within the past few years by applicants and the USAEC regulatory staff, together with the USAEC Advisory Committee on Reactor Safeguards, have resulted in a number of changes in plant design or operation. This paper discusses some of the most significant of these changes and the reasons for them. Examples are given that relate to site and environmental characteristics, reactor design, coolant-system design and performance, design of the containment and other structures, design and performance of engineered safety features, instrumentation and power-system design, and quality-assurance procedures. Although much progress has been made in reactor safety, more remains to be done. Experience with and analysis of actual systems uncover new safety issues and new aspects of old safety issues that require further study and resolution. Some of the most important such technical safety issues currently under consideration are discussed. One of the most challenging questions is the appropriate balancing of risk and benefit. For the safety reviewer, this means establishing acceptable risks for accidents and anticipated operational occurrences and therefore reliability criteria for protective systems and engineered safety features. Available technology and information are reviewed, and principles are stated for considering risk analysis in the absence of adequate data. The performance of protective systems and engineered safety features must be considered as well as their reliability. Protective systems whose performance is studied include the reactor-protection instrumentation system, the emergency-core-cooling system, systems to control combustible gas mixtures in the containment, and systems to control the radioactivity in liquid and gaseous effluents. It is becoming evident that human errors and common-mode failures of redundant elements may be the principal causes of protective-system failure. The use of various types of diversity is discussed.

DONNEES TECHNIQUES RELATIVES A LA SECURITE DES REACTEURS NUCLEAIRES DE GRANDE PUISSANCE.

Bien que les méthodes et les données fondamentales pour assurer la sécurité des réacteurs n'aient pas changé, une information considérable a été acquise grâce aux projets de recherche et de développement et au cours de l'analyse de la construction et du fonctionnement des réacteurs de puissance. Au cours de ces dernières années, les rapports sur la sécurité des réacteurs, établis par les équipes projetant la construction de nouveaux réacteurs, par les responsables de la réglementation à l'USAEC et par le Comité consultatif sur la sécurité des piles de l'USAEC, ont abouti à un grand nombre de changements dans la construction et l'exploitation des réacteurs. Le mémoire discute les changements les plus importants et les raisons qui les ont amenés. Des exemples sont donnés qui ont trait aux caractéristiques du site et de l'environnement, à la construction du réacteur, à la construction et au fonctionnement du système de refroidissement, à la construction de l'enceinte étanche et d'autres structures, à la construction et au fonctionnement des dispositifs de sécurité, à l'instrumentation, aux dispositifs de puissance, et aux méthodes pour maintenir la qualité du matériel. Bien que la sécurité des réacteurs ait considérablement progressé, il reste encore beaucoup à faire. L'un des problèmes les plus épineux est d'équilibrer les risques et les avantages. Pour l'examen de la sécurité cela signifie qu'il faut définir des risques acceptables d'accident et autres événements pouvant se produire au cours de l'exploitation, et par conséquent des critères de sécurité de fonctionnement pour les dispositifs de protection et les systèmes de sécurité. Les informations et techniques disponibles sont examinées et les principes d'une analyse des risques, en l'absence de données suffisantes, sont établis. Le fonctionnement des systèmes de protection et des dispositifs de sécurité doit être examiné, ainsi que leur sûreté. Les systèmes de protection dont le fonctionnement est étudié comprennent l'instrumentation pour la protection des réacteurs, les systèmes de refroidissement du cœur en cas d'accident, les dispositifs pour le contrôle des mélanges gazeux inflammables dans l'enceinte étanche et les systèmes de

contrôle de la radioactivité des effluents gazeux et liquides. Il devient évident que les erreurs humaines et les pannes ordinaires des dispositifs sont les causes principales de faillite du système de sécurité. L'utilisation de plusieurs types de dispositifs est discutée.

ТЕХНИЧЕСКИЕ АСПЕКТЫ БЕЗОПАСНОСТИ КРУПНЫХ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ.

Несмотря на то что основные требования и средства по обеспечению безопасности реактора остаются неизменными, осуществление исследовательских и опытно-конструкторских проектов, а также проведение анализа и приобретения опыта в области проектирования, строительства и эксплуатации энергетических реакторов позволили узнать много нового. Результатом изучения вопросов безопасности, проводившимся в течение последних нескольких лет предприятиями и законодательными органами КАЭ США, совместно с Консультативным комитетом КАЭ США по гарантиям безопасности реакторов, явился ряд изменений в конструкции и в порядке эксплуатации установок. В докладе рассматриваются некоторые из наиболее существенных изменений и причины их внедрения. Приводятся примеры, касающиеся характеристик площадок для строительства и окружающей среды, конструкции реактора, конструкции и работы системы теплоносителя, конструкции защитной оболочки реактора и других конструкций, особенностей конструкций и безопасности их работы, конструкции контрольно-измерительной аппаратуры и энергосиловой системы, а также методов проверки качества. Несмотря на то что уже достигнут значительный прогресс в области безопасности реакторов, многое еще остается сделать. Опыт работы и анализ существующих систем раскрывают новые проблемы обеспечения безопасности и новые аспекты старых направлений в этой области, которые требуют своего дальнейшего изучения и решения. В докладе затрагивается ряд наиболее важных технических аспектов безопасности, находящихся в настоящее время в стадии рассмотрения. Одним из самых важных вопросов является соответствующее сопоставление риска и выгоды. Для человека, занимающегося вопросами безопасности, это означает установление приемлемых рисков в случае аварий и возможных эксплуатационных случайностей, а потому и критериев надежности защитных систем и технических аспектов безопасности. Рассматривается современная технология и имеющаяся информация, излагаются принципы анализа риска в случае отсутствия необходимых данных. Работа защитных систем и технические аспекты безопасности должны рассматриваться вместе с вопросами обеспечения их надежности. Изучаемые защитные системы включают систему контрольно-измерительной аппаратуры по защите реактора, систему аварийного расхолаживания активной зоны реактора, системы для контроля образований горючих газовых смесей под защитной оболочкой реактора, а также системы для контроля радиоактивности в жидких и газовых отходах. Становится очевидным, что ошибки, допускаемые человеком, и общие неполадки у многочисленных элементов могут быть главными причинами выхода из строя системы защиты. Рассматриваются различные виды аварийных ситуаций.

PROBLEMAS TECNICOS DE SEGURIDAD EN GRANDES CENTRALES NUCLEARES.

Aunque las condiciones fundamentales y los medios para lograr la seguridad de los reactores permanecen inalterables, se ha progresado mucho gracias a las actividades de investigación y desarrollo y al análisis y experiencia en materia de proyecto, construcción y funcionamiento de reactores de potencia. Los estudios de seguridad realizados durante los últimos años, por los solicitantes de autorizaciones y el personal competente de la USAEC, juntamente con el Comité Consultivo sobre Salvaguardias de Reactores de la USAEC, han dado por resultado una serie de modificaciones en la concepción técnica de las plantas o en su funcionamiento. En la memoria se examinan algunas de las modificaciones más significativas y las razones a que obedecen. Se exponen ejemplos referentes a las características del emplazamiento y del medio ambiente, a las particularidades del reactor, a las características y eficacia del sistema de refrigeración, a las particularidades de la contención y de otras estructuras, al tipo y eficacia de los dispositivos mecánicos de seguridad, a las características del instrumental y de la red eléctrica y a las normas para garantizar la calidad. Aunque se ha avanzado mucho en la seguridad de los reactores, es todavía más lo que queda por hacer. La experiencia adquirida con los sistemas actuales y el análisis de dichos sistemas suscitan nuevos problemas de seguridad y aspectos inéditos de problemas que requieren más amplio estudio y solución. Se pasa revista a algunos de los más importantes problemas técnicos de seguridad actualmente en estudio. Una de las cuestiones más difíciles es conseguir un equilibrio adecuado entre riesgos y beneficios. Para el experto en cuestiones de seguridad, esto significa fijar los riesgos admisibles de accidentes e incidencias previsibles de funcionamiento y, por tanto, establecer los criterios de fiabilidad aplicables a los sistemas de protección y a los dispositivos técnicos de seguridad. Se pasa revista a los conocimientos e información disponibles y se exponen principios para el análisis de riesgos, cuando se carece de datos adecuados. Hay que tener en cuenta la eficacia de los sistemas protectores y de los dispositivos técnicos de seguridad, así como su regularidad funcional. Se estudia la eficacia de los siguientes sistemas protectores: instrumental de protección del reactor, sistema de refrigeración de emergencia del núcleo, sistemas para controlar las mezclas gaseosas combustibles en la

contención y sistemas para registrar la radiactividad de los efluentes líquidos y gaseosos. Resulta evidente que las causas principales de los fallos de los sistemas protectores son los errores humanos y las averías corrientes de los elementos accesorios. Se estudian, también, varios tipos de diversidad.

1. INTRODUCTION

Reactor safety - the avoidance of undue risk to the public health and safety from the operation of nuclear reactors - involves both technical and nontechnical issues. Setting the goals, numerical or otherwise, for acceptable risk and safety is primarily a social and political task. Translating these goals into criteria against which to evaluate proposals for the design and operation of reactors requires social and technical decisions arrived at with great difficulty. Most of the technical effort in reactor safety evaluation is devoted to a more mundane task: comparison of specific reactors against such criteria. This technical effort is performed principally by the plant owner and operator himself, or through his contractors, since the owner is responsible for the health and safety of the public as well as his employees.

In the United States, as in other countries with nuclear reactors, additional responsibility for the health and safety of the public is held by an agency of government. Such responsibility necessarily involves safety evaluations of the same type--if not of the same scope--as those conducted by the owner. In addition, the government agency plays a major role in the development of the criteria referred to earlier. In this paper the authors discuss some of the more important and interesting technical issues that have come to the attention of the Advisory Committee on Reactor Safeguards and the AEC regulatory staff in the last few years.

Some of the issues discussed are considered to have been resolved; that is, both the criteria and acceptable design approaches to meet them are available. Ideally, experience and measurements will have confirmed the design, construction, and operation of the devices or features involved in such a "resolved issue". By no means all safety issues are in this happy state. Others remain, to a greater or lesser extent, short of complete resolution. Lack of resolution often occurs because of the inevitable time delay between identification of an issue and completion of the work needed to resolve it.

For some kinds of issues, complete resolution is not foreseeable as a practical matter. Such problems usually involve events, or combinations of events, of such low probability that thousands of reactor lifetimes would not be expected to encompass even one observed event. Some postulated occurrences used for reactor safety evaluation fall in this class. Thus, design, construction and operating experiences are not expected to confirm the resolution of this kind of issue.

Such rare postulated events--for example, rupture of a large primary system pipe, or failure of the control rods to insert when needed, or occurrence of a severe earthquake--require careful consideration in reactor safety evaluation even though knowledge about them is and must remain incomplete.

It is such issues as these that constitute the most challenging and difficult portions of safety evaluation. Decisions on such issues necessarily involve judgments that must be made without complete information - judgments, therefore, subject to change as additional knowledge is gained.

This paper discusses the general subject of reactor safety goals and criteria, some examples of technical safety issues that have been resolved and finally, some examples of technical safety issues that are unresolved.

2. REACTOR SAFETY GOALS AND CRITERIA

2.1 Safety Goals

The ultimate reactor safety goal is elimination of risk to the health and safety of the public. The principal potential hazard associated with operation of nuclear reactors lies in the possible release of the radioactivity of the heavy elements and fission products produced in the fuel. The safety goal, therefore, is prevention of the exposure of people to this radioactivity. This goal can be achieved with a high degree of assurance, but not perfectly, since the possibility of fuel failure and consequent release of some radioactivity to the environment cannot be precluded absolutely. The ultimate reactor safety goal is thus theoretically unattainable, and a certain residual risk is unavoidable.

The minimization of risk is a realistic, practical goal, but it introduces a dilemma: What is an acceptably low level of risk? What is the basis for choosing it? Minimization is not enough. By choosing operating parameters with greater conservatism, adding more safety features, using more remote sites, one can obtain at least the illusion of greater safety. This process has no end except diminishing returns, or at worst a decrease in overall safety as a result of overcomplication. What is needed is some delineation of what is enough. The necessarily imperfect goal must be translated into a guide to the designer and the person charged with safety.

The decisions as to what risk is acceptable -- as defined piecemeal in criteria, standards, codes, guides -- have, in the past, been guided largely by experience and judgment. For some aspects of reactor safety, formulation of criteria in this way is a relatively tractable task. For the rare postulated events discussed previously, however, this method of establishing criteria is difficult for two reasons. First, the probability - expected occurrence rate - of these events is far too small to permit the accumulation of experience necessary to proceed in a meaningful way. Second, the possible consequences of some postulated failures are so severe that the whole procedure is not applicable. Accumulation of experience - even if it were possible - is unthinkable for highly unlikely events that might kill or injure a very large number of people. Since the acceptability of nuclear reactors, or anything else, depends on the fact that the unthinkable events are very rare indeed, an alternative must be found for dealing with such events in the establishment of criteria.

2.2 Risk Analysis

A seemingly most attractive method for assessing reactor safety goals and criteria is based on a numerical quantification of risk. In its simplest form, the theory defines risk as the product of the probability of occurrence of an event times the consequence of that event, also suitably quantified. Similar, more complicated methods also have been proposed. The total risk is the sum of the calculated risks for everything that can happen. This approach and others of this nature are not the infinite tasks they would at first sight appear to be. A few events are found to dominate the analysis. For plants to be acceptable, extreme cases of high probability or (at the other end of the scale) serious consequences must be found to involve a low risk.

The potential advantages of such an approach are obvious. The evaluation automatically emphasizes events that contribute significantly to the risk, which is where the emphasis belongs. No decisions need be made about credibility, since "incredible" events eliminate themselves from consideration by their low risks arising from their low probabilities.

Closely associated with risk evaluation is the technology of reliability analysis, used to predict the required probabilities of occurrence. It is in this area that problems arise, in the authors' opinion, which decrease the utility of risk calculations in reactor safety evaluations. For all its proved worth in many areas of technology, application of probability analysis to the type of rare events considered in nuclear safety involves the following uncertainties: (a) There is little or no concrete evidence that the events under consideration obey the laws of probability that underlie the theory. There is no assurance that accident events are random in time or independent of each other. The intellectually satisfying idea of reliability analysis is no more than a hypothesis for these events; (b) Even if the framework of the theory were correct, the values of the parameters are largely unknown.

The method now generally used to predict these probabilities -- cascading of probabilities of the individual "failures" that make up the event -- is known to be inadequate. The serious or potentially serious events that have occurred have been characterized by concurrent failures, usually interdependent or causally related. Thus the theory's assumption of independence of failures has not been borne out by experience.

Despite these shortcomings, reliability analysis and, to a lesser extent, risk calculation techniques are used in reactor safety evaluation in the United States. In some areas, such as meteorology, there is a substantial body of knowledge to serve as a basis for the calculations; these results are thought to approximate reality. Where the basis is less firm, the calculations are nonetheless useful in making comparisons on a relative basis, as opposed to an absolute basis, thereby directing attention to the more important considerations.

2.3 Design Basis Accidents

The principal tool used in the United States for reactor safety evaluation is not risk calculation but analysis of a

spectrum of design basis accidents. The response of the reactor plant to each of these postulated events is predicted and the radiological consequences are compared with arbitrary criteria. Two such criteria are used: one for events expected to occur frequently during the reactor lifetime, the other for very unlikely events. Subdivision of classes of postulated events is surely possible and has been proposed; the present state of knowledge appears inadequate to the authors to justify quantification of intermediate events, but qualitative comparisons appear to be meaningful.

It is evident even from this brief discussion that despite the explicit rejection of quantitative risk analysis for safety evaluation, notions of probability do play an important role. A more careful investigation suggests not two but four classifications of events:

- Class 1. Expected occurrences whose effects or consequences must be negligible.
- Class 2. Occurrences whose frequency is intermediate between Class 1 and Class 3, that are only beginning to be defined and treated separately.
- Class 3. Unlikely severe accidents used for design bases.
- Class 4. "Incredible" events against whose consequences no protection is required.

The classification is based on probability, even though the values of the probabilities are not well known. If there is any doubt, an event can be placed in a more probable classification to assure that the evaluation is on the conservative side of reality. The classification scheme is seen to be a crude approximation to risk analysis.

2.4 Defense in Depth

The concept of defense in depth is another way of expressing the foregoing ideas. Reactor designs are required to include a number of barriers between people and the radioactivity contained in the fuel. In water reactors these barriers are the fuel matrix itself, the fuel cladding, the primary system pressure boundary, and the containment. Expected operating transients and other Class 1 events are required not to jeopardize any of the barriers. Any failures in the fuel or its cladding would release radioactivity to the primary system, from which only small leakage would occur; a radioactivity cleanup system would prevent any significant release of radioactivity to the environment. For Class 2 events, and most Class 3 events, engineered safety features protect the primary system, although some of these events could lead to radioactivity releases. Severe postulated breaches of the primary system (the most severe Class 3 events) could lead to rupture of the fuel cladding and release of radioactivity from the fuel and through the breach into the containment; emergency core cooling systems and other engineered safety features would limit the radiological consequences to persons off site.

Class 1 events are expected to happen, so that their consequences must be negligible or nearly so. Practically, the design of features to cope with Class 1 events will be determined by economic considerations. For Class 2 and Class 3 events, the acceptable consequences are progressively more severe commensurate with their lower probabilities. Class 4 events have such low probability that their consequences need not be considered.

The principal defense against accidents is prevention. All structures, systems, and components important to safety must be designed, built, and operated so that the probability of occurrence of an accident is very small. The key to achievement of this objective is an effective quality assurance program, discussed at length elsewhere.¹ However excellent the quality assurance program, it must be acknowledged to be imperfect. Protective systems are installed therefore to deal with such transients and failures as may occur despite all that is done to prevent them. A third echelon of the defense in depth is the engineered safety features designed to cope with unlikely failures that go beyond the capabilities of the accident prevention and protective systems, as well as highly unlikely failures of the other defenses themselves. The designs of engineered safety features are evaluated to provide assurance that they will function properly under accident conditions. Each line of defense must be well designed and executed for effective implementation of the defense-in-depth concept. For example, system performance is evaluated assuming a failure of any single active component in any engineered safety feature.

In principle, defense-in-depth can be proliferated endlessly, analogous to the possible proliferation of design basis accidents. Diminishing returns from such proliferation dictate establishment of a limit to the required defense-in-depth, again analogous to the distinction between "credible" (Class 3) and "incredible" (Class 4) events. This limit, expressed as either a requirement for depth of defense or an array of credible events for which protection is required, is one of the most difficult technical safety issues to resolve. As usual, the lack of knowledge regarding probabilities is responsible for the difficulty. Judgment is rendered on an inadequate basis, and therefore is subject to change as additional knowledge is gained.

3. SOME TECHNICAL SAFETY ISSUES THAT HAVE BEEN RESOLVED

Listed in this section are examples of individual actions that have been taken on technical safety issues. Many of these actions were taken to achieve conformity or compliance with existing criteria and guides of the AEC; many were the result of decisions made on matters that arose in the context of individual cases. These examples are provided to describe the kinds of actions that have been taken, not to comment on or analyze the bases for taking such actions.

¹ See, for example, BECK, C. K., Role of the United States Atomic Energy Commission in control and licensing of nuclear power reactors, Nucl. Engng (Nov. 1970).

3.1 Site and Environmental Matters

Design specifications for facility structures and equipment to accommodate higher earthquake acceleration values than originally proposed have been required in many instances.

Design modifications to provide higher flood protection levels (including hurricane, storm surge, seiche and tsunami protection), than originally proposed have been required in many instances.

Protection of vital structures and equipment against tornadoes is required. The design bases for spent fuel pools have been modified in a number of cases to protect against loss of water, from pool damage, and release of fission products from fuel damage as a result of missiles propelled by tornadic winds.

For one plant site located near an airport, design modifications were required to protect against impact and fire effects of aircraft crashes.

Vent stacks have been required to be located away from containment buildings to avoid damage from a falling stack where the stacks were not designed to withstand tornadic winds.

Installation of strong motion seismographs is now a requirement for all plants.

Extensive soil and rock drilling programs have in some cases shown the need for remedial action for solution cavities, for relocation or redesign of structures, or for replacement and compaction of soil to provide suitable foundation conditions.

Design changes, such as relocation of equipment, have been required to accommodate the effects of the failure of upstream dams.

Instrumentation on towers has been required onsite to obtain local meteorological data to support adequately the assumptions made for atmospheric dispersion of radioactivity.

Environmental monitoring programs have been required to be augmented.

3.2 Reactor Design

Part-length control rods and revised instrumentation were provided to assure the capability for observing and controlling the potential effects of xenon oscillations and other power distribution anomalies.

Fixed reactivity shims were provided in the reactor core to reduce the initial value of the moderator coefficient of reactivity from a positive value to zero, so as to reduce the reactivity that would be inserted during a loss-of-coolant accident with the positive coefficient.

The capability to include fixed, incore flux detectors has been required where this capability did not exist originally.

Vibration test programs currently are required during initial cold and hot testing at the startup phase to assure acceptable performance of reactor internals and to monitor for loose parts.

3.3 Reactor Coolant System Design and Performance

Additional pipe hangers were installed as a result of a revised dynamic analysis of system piping.

Design of main steam lines was required to include a simplified dynamic analysis and improved quality control on field welds.

Stress analysis and inspection of primary coolant pump flywheels during fabrication and the capability for inspection during the service life of the flywheels is now required for all plants.

Flow restrictions have been required in main steam lines to reduce the reactivity transient following a postulated steam line break.

Material samples for monitoring the integrity of sensitized stainless steel within the primary coolant system have been required.

The design bases for reactor internals inside the reactor vessel were modified to include an allowance for blowdown forces following a loss-of-coolant accident.

The sensitivity and reliability of detection of coolant leaking from the primary system and from emergency systems were required to be improved.

Additional pipe restraints and inservice inspection have been required in the vicinity of steam generators to reduce the probability of a primary system pipe break causing a steam generator failure.

Additional restraints have been required on primary system piping to prevent the possibility of pipe whip, following a break, rupturing the containment boundary or causing other piping failures.

Replacement, or other measures to minimize the potential for cracking of furnace-sensitized stainless steel have been required in several plants.

3.4 Containment and Structures

Diagonal reinforcement for the primary containment building was required, to accommodate seismic shear forces.

Aluminum components used inside the containment building were replaced to the extent possible with steel to minimize the amount of hydrogen that might be generated from corrosion of the aluminum by the chemical solution of the containment spray system.

Improved containment building penetration design details, including leak testing ability, were required for a number of plants.

Additional protection of fuel storage pools is being required to protect against the consequences of dropping a shipping cask into the pool.

Control room shielding was required to be increased in thickness to allow normal occupancy following a postulated loss-of-coolant accident.

A fresh water storage tank was relocated since failure at its original location could have caused the failure of two diesel generators.

The intake channel and intake structure for the condenser cooling water were modified to withstand the design basis earthquake and to provide emergency cooling at minimum water level.

A containment building design was modified so that containment prestressing cable anchors would be easily accessible for inspection during operation.

All portions of the post-accident decay heat removal systems are being required to be designed to seismic Class 1 specifications.

The gaseous waste storage tanks are being required to be designed to seismic Class 1 specifications.

The response spectrum for seismic design was required to be revised to provide a more conservative design.

The design of fuel storage building was required to be revised to include provisions for controlling leakage of gaseous activity and for charcoal filtration of gaseous effluent.

Provisions are now required to deal, without the necessity of venting, with potential combustible gas mixtures generated, following a loss-of-coolant accident, by metal-water reactions in the core, radiolysis of emergency core coolant, and corrosion in the containment.

3.5 Engineered Safety Features

The designs of emergency core cooling systems and vital heat removal systems have been required to be revised to increase reliability and margin and to meet active and passive failure criteria.

Emergency Core Cooling System pump motors were required to be protected against the spray of the containment cooling system.

Chemical sprays or filters (charcoal and particulate) are being required to reduce iodine inventory after a loss-of-coolant accident to meet reduction factors required by conservative assumptions on fission product releases and meteorology.

Redundant diesel generators have been required in a number of cases to achieve adequate reliability of onsite power systems.

Rooms housing critical emergency core cooling components have been required to be sealed or separated to prevent flooding of all equipment as a result of a single failure.

Tests of vital equipment have been required under conditions simulating those attending an accident.

Filters were required to be added to the auxiliary building ventilation systems for a number of plants to mitigate the effects of a fuel handling accident.

The design was modified in a number of cases to include automatic isolation of the main condenser mechanical vacuum pump and gland seal exhausters to reduce potential doses from a rod-drop accident.

The amount of fuel for emergency diesel generators stored onsite is required to be sufficient to allow operation of engineered safety features for one week.

The containment air cooler design was required to be modified to allow detection and isolation of failure of the service water coolant lines.

Radiation detectors were required to be placed adjacent to the spent fuel pool to provide quicker detection of fission product release and resultant isolation of the reactor building in the event of leakage.

The control room ventilation system was required to close automatically on detection of radioactivity in inlet air.

Electrical heat tracing was required to be provided on the standby liquid poison control injection line (boron solution).

Physical separation and redundancy of standby gas treatment systems were required.

More conservative limits were required on the stresses, strains and deformations permitted in Class 1 systems, structures and components under seismic and accident conditions in a number of plants.

Additional equipment has been required to reduce the activity level in off-gas releases.

The design leakage rate for several containment buildings was required to be reduced.

Installation of independent overspeed protection for the turbine generator has been required for several plants.

3.6 Instrumentation and Power Systems

Changes have been required in a number of plants to achieve separation of cables servicing redundant control and protection instrumentation.

Changes have been required to improve the testability of vital instrumentation.

Switching of equipment loads between emergency diesel generators was changed from automatic to manual.

Incorporation of the split bus arrangement for emergency power supply has been required for a number of plants, resulting in the addition of another diesel generator and redesign of distribution systems.

Circuitry for a flow-biased, flux scram was required to be installed and to meet protection system standards for a number of reactor plants.

The circuitry for the rod block monitor system and auto-relief system were required to meet protection system standards for several reactor plants.

The control rod position readout system was required to be modified to give indication when rods are off their demanded position.

Instrumentation is now required to meet seismic design standards in all plants.

A redundant station battery was required.

The ability to perform a hot shutdown and the potential capability to perform a cold shutdown from outside the control room is currently required.

Diverse signals are required for actuation of the emergency core cooling systems.

Means for direct measurement of primary system coolant flow rate were required.

An alternate circuit to provide offsite power to the emergency buses was required to cope with postulated failure of the startup transformer of several plants.

Actuation instruments for engineered safety features are required to meet protection system standards.

Control systems for emergency power were required to be redesigned to meet protection system standards.

3.7 Miscellaneous

Improved development of Quality Assurance procedures and organization changes to implement these quality assurance programs have been required in many cases.

Design changes have been required in radwaste systems to reduce normal operational releases.

Redesign of fire protection systems has been required so that failure of seismic Class 2 portions of the systems will not prevent the functioning of seismic Class 1 equipment.

Interlocks and procedural restrictions have been required with respect to the crane which hoists to minimize the potential for cash drop which could damage fuel or the fuel pool.

Post-operational integrated leak rate and strength tests of the containment building are required.

Changes have been required in the operating organizations for a number of plants to include an additional supervisor and the number of licensed operators.

4. SOME TECHNICAL SAFETY ISSUES THAT HAVE NOT BEEN RESOLVED

Some examples of technical issues that are not resolved are discussed in this section. Adequate assurance of safety does not require complete resolution of these matters (see Section 1.) provided approximate conservative assumptions and procedures are used in areas where knowledge is limited.

4.1 Thermal Shock from Emergency Core Cooling

As an example of an "unresolved" technical safety issue, consider the consequences of injection of emergency core cooling water into the reactor vessel. The reactor vessel, initially at 300°C, is sprayed or flooded with cooling water initially at 25°C. The resulting thermal stress in the vessel wall can be calculated; it is not difficult to show that a perfect vessel will not be damaged. But real vessels are imperfect in at least two ways: (a) flaws smaller than some acceptance limit remain after manufacture and inspection; and (b) the properties of the material change during life because of the neutron irradiation. Therefore, the stress calculation must be performed with some conservatively hypothesized flaw present in an embrittled steel.

The only way such calculations can be made presently is ultra-conservatively, since the fracture toughness properties of thick (15-25 cm) sections of irradiated steel are not well known. These over-conservative predictions show that following long irradiation such an event might cause the vessel to crack severely. A large-scale program is underway to develop both the analytical tools and the experimental information needed to reduce the conservatism now required, but several years must elapse before this information will be available.

Today's vessels are not highly irradiated, because today's large reactors are young. So, one can put off the determination of whether something must be done about thermal shock until the information is available. If the present conjecture is disproved, and remedial measures become needed, the vessels can be annealed in place as their neutron doses reach the point where this is indicated.

Thus this problem has -- and needs-- no definitive solution for some time to come, and an appropriate program is underway to resolve the issue when needed.

4.2 Power and Peaking Limits

Another technical issue, that will never be "finally" resolved, relates to the power rating of a reactor core. The early reactors were rated very conservatively. After experience had been gained at these levels, development programs were undertaken to show, theoretically and experimentally, that higher ratings were justified. In many cases, the increased power and power density, as compared with those of previously approved reactors, have been justified on the basis that the safety margins previously evaluated have been found to be larger than originally believed. This reevaluation is usually based on measurements in the newer operating reactors, or more sophisticated calculational techniques, or the data from the development program -- in some cases on all of these things. In many instances, the new information and techniques do indeed show that the improved performance leaves margins not significantly different from those formerly thought to exist. However, it is also undeniably true that the actual safety margin will have been diminished by the uprating.

If the power increase is justified by leaving the peak power density unchanged, but running the rest of the core at higher power density (improved flattening), then the issue is rather different. The hottest part of the core is demonstrably no hotter than before. The rest of the core, though hotter than before, is still cooler -- has a greater safety margin -- than the hot spot. But any accident involving overheating will involve a larger portion of the new core than the old one, because a greater fraction of the new core runs hotter, even though the hot spot does not.

What is an appropriate limit for this procedure? Economy dictates that the core heat generation should be as nearly uniform as practical, so long as the result is safe. So far, power distributions are not so flat as to give serious concern, but a limit of flattening has not been determined. Further study is needed on the flattening question so that development of higher rated, more economical cores will not be inhibited by lack of knowledge of this safety issue.

4.3 Common-Mode Failures in Protective Systems

Protective systems and engineered safety features are composed of redundant components, arranged so that no single failure will result in system failure. Yet, when the system failures that do occur are studied, it is discovered that the majority of such events involve multiple failures, for which the probability is calculated to be very low. The actual failures are found to involve not the concurrent independent component failures usually considered in probability calculations, but causally related failures from design error, exposure to hostile environment, or human error. These common-mode, or systematic, failures are distinguished from the random failures usually considered.

Redundancy offers little defense against common-mode failures. The only technique presently known to reduce their probability of occurrence is diversity; i.e., the ability to perform the function in different ways. For the reactor protection system, diversity can be applied by monitoring different process variables

to detect a given condition (signal diversity), by using different types of equipment to perform the same function (equipment diversity), and by employing different modes of corrective action (actuator diversity).

The study of these problems is in its infancy. A few incidents have been analyzed; a few protection systems have been studied systematically to identify potential common-mode failures; a few examples of signal diversity have been designed. The amount of improvement in reliability attainable by the use of diversity is not known, since no systems are yet known to have been designed in the United States with the full deliberate application of the three types of diversity. Moreover, the amount of improvement required, if any, is not known because the reliability of present systems is not known.

The controlling factor in requirements for protection-system reliability is the need for protection against certain anticipated operational occurrences. Examples of such occurrences are turbine trip for the boiling-water reactor and loss of power to the main circulating pumps for the pressurized water reactor. Since such events are expected to occur with a frequency of approximately once every few years, any events in this class for which failure of protective action would have unacceptable consequences will require a very high protection-system reliability in order to keep acceptably low the probability of an unprotected accident.

Study is continuing on both the probabilities of the various failures and occurrences and on their potential consequences.

4.4 Probability of Environmental Events

One of the design criteria for safety of a power reactor is that it be able to withstand the most severe earthquake, hurricane, flood or tornado that reasonably can be postulated. Clearly, each site must be evaluated in these regards. The difficulty in establishing the criteria for a specific site occurs because the recurrence interval of such severe events is very large -- thousands of years or longer. In the United States, this is longer than any historical record, and the nonhistorical record is often hard to read. The criteria for a given site therefore must be based on the best judgment of experts in seismology, meteorology, and so forth, based on frequency-severity theories necessarily unchecked in the region of concern. The result is that occasionally a site must be re-evaluated in the light of new information and the criteria revised (-- always upward, it seems. In the present state of knowledge, this appears to be unavoidable).

4.5 Performance of Emergency Core Cooling Systems

The loss-of-coolant accident (LOCA) is a Class 3 postulated improbable accident (see Section 2.3); the emergency core cooling system (ECCS) is one of the engineered safety features installed to mitigate its consequences.

The ECCS is required to be designed to do the following: (a) to limit the peak fuel cladding temperature to a safe value, currently taken to be 1260°C, (b) to limit the chemical reaction between the cladding and water to involve less than 1% of the total amount of cladding, (c) to terminate the temperature transient of the cladding so that the core geometry permits adequate cooling and before the cladding is so embrittled as to fail upon quenching, and (d) to reduce the core temperature and remove decay heat for an extended period of time.

As a result of an extensive review in 1966, programs were initiated by the Atomic Energy Commission and the nuclear industry leading to improvement in primary system integrity, development of improved analytical methods for predicting ECCS performance, and acquisition of experimental information to support and confirm the analyses.

The experimental and analytical programs are described in a companion paper.² The related processes of development of analytical tools and confirmatory experimentation are expected to continue. However, use of the new, more sophisticated, analysis techniques for evaluation of ECCS performance in reactor systems has begun. In view of the large amount of new information available, the Atomic Energy Commission again conducted, in 1971, a review of the current state of ECCS technology.

Ideally, one would have available analysis methods capable of detailed realistic prediction of all phenomena known or suspected to occur during a LOCA, supported in every aspect by definitive experiments directly applicable to the LOCA. In the absence of such perfection, adequate assurance of safety is obtained from an appropriately conservative analysis based on available experimental information. In areas of incomplete knowledge, conservative assumptions or procedures are applied. As further experimental information or improved calculational techniques become available, the conservatisms previously imposed can be reevaluated and a more realistic approach taken, justified by the improvement in knowledge.

The result of the 1971 review was the development of interim evaluation models consisting of computer codes, each together with its set of suitably conservative assumptions and procedures, to be applied to each plant.

² BROCKETT, G.F. et al., these Proceedings, Paper 39, Vol.3.

LA SEGURIDAD NUCLEAR EN ESPAÑA

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Abstract—Résumé—Аннотация—Resumen

NUCLEAR SAFETY IN SPAIN.

The paper describes the basic principles followed in Spain to assure that siting, design, construction and operation of nuclear installations or installations where radioactive materials are handled is done without undue risk. The safe transport of radioactive materials and nuclear fuels is also included. The 25/1964 law on nuclear energy establishes these basic principles, and also regulates the administrative rules to be followed in requesting and granting licenses. The basic principles of the law are further developed in the corresponding regulations on application.

After describing these principles, the paper deals with the technical documents which must be presented to the authorities in requesting a license. In licensing nuclear installations, four steps have been defined which can be identified with: site selection, construction, nuclear testing and operation. Other details of Spanish practices on nuclear safety are presented.

The report also presents the principles to be followed in evaluating the safety documents, among which are the safety analyses, emergency manuals, limits and conditions for operation, periodic inspection and testing, etc. Finally, the experience gained with the nuclear installations in Spain, among which there are a PWR, a BWR and a gas-cooled reactor, is treated. This experience also includes, to a lesser extent, installations for the handling of radioactive materials and the transportation of radioactive materials.

LA SECURITE NUCLEAIRE EN ESPAGNE.

On expose et on analyse les principes fondamentaux appliqués en Espagne pour garantir que l'emplacement, le projet, la construction et l'exploitation des installations nucléaires et radioactives n'occasionnent aucun risque indu. Le transport sans danger de matières radioactives et de combustibles nucléaires est également discuté. La loi 25/1964 sur l'énergie nucléaire en établit les principes de base et spécifie également les formalités administratives à observer pour la demande et l'obtention des différentes autorisations. Les principes de base de la loi sont développés dans des règlements d'application.

Le mémoire développe ces principes et décrit les documents techniques qui doivent accompagner les différentes demandes d'autorisation. Dans le cas des installations nucléaires, on distingue quatre phases dans le régime des autorisations, soit le choix du site, le début de la construction, la vérification sur le plan nucléaire, et l'exploitation définitive. D'autres particularités concernant les pratiques suivies en Espagne dans l'application de la sécurité nucléaire sont également exposées.

On décrit les principes de base à suivre pour l'évaluation des directives données dans des documents tels que les notes d'information sur la sécurité, les manuels pour les cas d'urgence, les limites et obligations, le programme d'inspections ainsi que les confirmations périodiques et règlements d'exploitation. Enfin on expose l'expérience acquise dans les centrales nucléaires du pays, qui comprennent des réacteurs à eau sous pression, à eau bouillante et au graphite-gaz. L'expérience porte également, à un degré moindre, sur les installations radioactives et le transport de matières radioactives.

ЯДЕРНАЯ БЕЗОПАСНОСТЬ В ИСПАНИИ.

В докладе описываются основные принципы безопасности, принятые в Испании при выборе строительной площадки, а также при проектировании, сооружении и эксплуатации ядерных установок или установок, в которых применяются радиоактивные вещества, с целью предотвратить ненужный риск. Включены также вопросы безопасности при перевозке радиоактивных веществ и ядерного топлива. Основные принципы установлены Законом №25/1964 о ядерной энергии, который также определяет административные процедуры подачи заявок и получения лицензий. Установленные законом основные принципы разработаны в соответствии с действующими инструкциями.

После изложения указанных принципов в докладе рассматриваются технические документы, которые необходимо представлять в правительственные органы для получения лицензий. Определены четыре этапа получения лицензий на ядерные установки, включая выбор площадки для строительства, сооружение, ядерно-физические испытания и эксплуатацию. Рассматриваются другие особенности практики обеспечения ядерной безопасности в Испании.

В докладе излагаются также принципы оценки документов по вопросам безопасности, которыми являются аналитические отчеты по вопросам безопасности, справочники по аварийным ситуациям, перечни ограничений и условий эксплуатации, периодического инспектирования и проведения испытаний и т. д. В конце доклада дается описание опыта, приобретенного при работе на ядерных установках в Испании, включая такие установки, как реакторы PWR и BWR, а также реактор с газовым теплоносителем. В несколько меньшем объеме обсуждаются вопросы, касающиеся установок для работы с радиоактивными веществами и перевозки радиоактивных веществ.

LA SEGURIDAD NUCLEAR EN ESPAÑA

Se exponen y analizan los principios fundamentales seguidos en España para garantizar que la ubicación, proyecto, construcción y explotación de las instalaciones nucleares y radiactivas se lleva a cabo sin riesgo indebido. Se incluye también el transporte sin riesgos de materiales radiactivos y combustibles nucleares. La Ley 25/1964 sobre energía nuclear establece los principios básicos que deben seguirse a tal respecto, también especifica los trámites administrativos a seguir en la solicitud y concesión de las distintas autorizaciones. Los principios básicos de la Ley se desarrollan en Reglamentos de aplicación.

La memoria desarrolla tales principios y describe los documentos técnicos que deben acompañar a las distintas solicitudes de autorización. En el caso de instalaciones nucleares se distinguen cuatro fases en el régimen de autorizaciones, correspondientes respectivamente a la selección del emplazamiento, el comienzo de la construcción, la verificación nuclear y la explotación definitiva. También se exponen otras peculiaridades de las prácticas seguidas en España en la aplicación de la seguridad nuclear.

Se describen los principios básicos que se siguen en la evaluación de los llamados documentos preceptivos, entre los que se incluyen los Informes de Seguridad, Manuales de Emergencia, Límites y Condiciones, Programa de Inspecciones y Comprobaciones Periódicas y Reglamentos de Explotación. Finalmente se expone la experiencia adquirida con las centrales nucleares del país, que incluyen instalaciones con reactor de agua a presión, de agua en ebullición y de grafito gas. La experiencia incluye, en menor grado, a las instalaciones radiactivas y al transporte de materiales radiactivos.

1. - Bases legales

Cuando en España se empezaron a vislumbrar las posibilidades de la energía nuclear se inició la tarea de constituir el marco jurídico adecuado para llevar a cabo las actividades que con ella se relacionaban. Se crean las instituciones necesarias para coordinar los esfuerzos en esta materia y poco a poco un cuerpo legislativo irá trazando el camino que conduzca al fin propuesto: el fomento de las aplicaciones pacíficas de la energía nuclear y la protección de la comunidad nacional contra los riesgos que de ella se derivan.

La labor comenzada varios años antes culmina en 1964 con la publicación de la Ley 25/1964, de 29 de abril, sobre Energía Nuclear. La Ley se concibe como un amplio conjunto de disposiciones que regulan diversas actividades en el campo de la energía nuclear y en todos sus aspectos jurídicos. Por esta razón se incluyen en el ámbito de la Ley no sólo las instalaciones nucleares, definidas como lo habían hecho los Convenios de París y de Viena, sino también las instalaciones radiactivas, la minería y el transporte de materiales radiactivos, buques y aeronaves nucleares. Pero también la Ley contiene las disposiciones referentes a los organismos gubernamentales a quienes se encomienda su ejecución, y se dan

normas sobre el personal que trabaja en estas actividades y se ocupa de la investigación y enseñanza nuclear. En cuanto al carácter de sus normas puede decirse que abarcan prácticamente todas las ramas de la ciencia jurídica. El derecho administrativo, el laboral, el derecho civil, el penal, etc., se hallan representados en esta Ley, que se convierte en la legislación especial de esta materia específica. Sin embargo, para llevar a cabo la regulación se evita la forma excesivamente detallada y rígida. Dado el contenido técnico que forzosamente deben tener muchos de sus preceptos y la continua evolución de la materia regulada, se elige una formulación amplia de los principios básicos que después habrán de desarrollarse en Reglamentos de aplicación, más fáciles de enmendar o sustituir a medida que la experiencia o el cambio en el hecho técnico lo aconsejen.

Esta idea del legislador se pone de relieve especialmente al observar las disposiciones de la Ley que se refieren a la seguridad nuclear y a la protección contra las radiaciones en las que, si bien se establece un sistema para su puesta en práctica y se dan las normas fundamentales del mismo, se ha omitido la regulación detallada para que, a la luz de la experiencia y ante las necesidades que vayan surgiendo, se publiquen las disposiciones complementarias.

En la actualidad, aunque los Reglamentos de aplicación previstos en la Ley se hallan en adelantada fase de elaboración, aún no han sido publicados. Por esta razón, ateniéndose al sistema establecido en la Ley y a las directrices que ésta marca, siempre teniendo presentes los objetivos de la legislación nuclear, diversas Resoluciones administrativas han ido desarrollando los criterios de seguridad para que las actividades nucleares se realicen sin riesgo indebido.

Los rasgos básicos del sistema legal en relación con la seguridad nuclear son: atribución a determinados órganos de la Administración de la competencia en esta materia; prohibición, refrendada por sanción penal, de efectuar cualquier actividad que implique posesión, utilización, manipulación o transporte de materiales radiactivos sin estar expresamente autorizado; obligación de cumplir las condiciones impuestas en la autorización.

2. - Organos Administrativos

Los órganos de la Administración directamente llamados por la Ley a ejercer el control de las actividades nucleares son el Ministerio de Industria y la Junta de Energía Nuclear.

El Ministerio de Industria tiene a su cargo todo lo relativo a autorizaciones, imposición de condiciones a las que se ha de ajustar la actividad autorizada y vigilancia del cumplimiento de estas condiciones, así como de las disposiciones de la legislación aplicable.

A la Junta de Energía Nuclear (J. E. N.), organismo autónomo que depende del Ministro de Industria pero goza de personalidad jurídica y

plena autonomía económica y administrativa, le está encomendado el informe preceptivo al Ministro de Industria en el trámite de las solicitudes de autorización y el análisis de los riesgos y la seguridad intrínseca, así como la inspección en este aspecto de las instalaciones nucleares y radiactivas.

De esta resumida exposición de las funciones de ambos Organismos se deduce que su misión respecto de cada actividad nuclear comienza con la solicitud de autorización y se extiende durante todo el ejercicio de la actividad.

3. - Normas básicas de seguridad

En efecto, el principio básico de la seguridad nuclear es la necesidad de autorización para realizar cualquier actividad que suponga la utilización, almacenamiento, manipulación o transporte de materiales radiactivos de cualquier clase.

Así, la Ley dispone que los materiales radiactivos y los combustibles nucleares no podrán ser almacenados ni utilizados dentro del territorio nacional por personas o entidades que no están autorizadas expresamente por el Ministerio de Industria, previo el preceptivo informe de la Junta de Energía Nuclear, exigiéndose estos mismos requisitos para su transferencia y reventa. En cuanto a las instalaciones nucleares y radiactivas, objeto primordial de las normas que aquí se analizan, se dispone que su construcción y montaje requiere autorización del Ministerio de Industria, siendo preceptivo el informe de la Junta de Energía Nuclear.

4. - Normas Administrativas

La Ley silencia el procedimiento utilizado para la tramitación de dichas autorizaciones, confiriendo al Ministerio de Industria la competencia para señalar los requisitos a cumplir y los trámites a seguir, y encargando a uno de sus departamentos, la Dirección General de la Energía y Combustibles, la tramitación. Por tanto, la práctica administrativa ha establecido los criterios y fases de que se compone el proceso de autorización desde la solicitud hasta el permiso definitivo para la puesta en marcha, según se expone más adelante, pero antes de entrar a examinarlo conviene hacer unas observaciones de carácter general.

El procedimiento se ha dividido en varias etapas, cada una de ellas sancionada por una Resolución administrativa donde se especifican las condiciones en que se concede y las actividades que pueden realizarse en cada fase. En la primera resolución se dispone la creación de un Comité de Coordinación compuesto de representantes de la Administración y del titular de la Autorización, que será el órgano encargado de velar por el cumplimiento de las condiciones impuestas. En caso de incumplimiento de sus condiciones, la misma Resolución suele hacer referencia al derecho de la Administración a dejar sin efecto la autorización, derecho incluido en la Ley entre las sanciones administrativas en que incurre la infracción de las normas sobre actividades nucleares.

Asimismo, la prevención de los posibles riesgos de daños nucleares estará siempre presente, pero también es de señalar que llegado un momento del procedimiento en que se faculte al titular a efectuar actividades que supongan la presencia en la instalación de combustibles nucleares o materiales radiactivos, se exige la cobertura del riesgo conforme se establece en la Ley y en el Reglamento de Cobertura del Riesgo de Daños Nucleares, para que, en caso de accidente, no queden las víctimas del mismo sin la oportuna reparación.

Terminado el proceso de autorización y facultado el titular de la misma para la puesta en marcha de la instalación se seguirá vigilando de forma continua el cumplimiento de las condiciones de seguridad para que su funcionamiento no dé lugar a daños. Por esta razón se establecen no sólo los requisitos que debe cumplir la instalación sino también las condiciones de idoneidad que habrá de reunir el personal que trabaje en estas instalaciones.

Por otra parte, tanto las personas profesionalmente expuestas como la población en general se encuentran protegidas en la legislación contra el riesgo creado por las radiaciones ionizantes, habiéndose adoptado una serie de medidas tendentes a evitar, en todo lo posible, los daños que éstas podrían ocasionar.

Estos son, en resumen, los principios jurídicos en que se basa la seguridad nuclear, su aplicación práctica, tal como se viene realizando en la actualidad, se efectúa en la forma que se expone en los apartados que siguen.

5. - Misiones de la Junta de Energía Nuclear en relación con la Seguridad Nuclear

La vigente Ley 25/1964 sobre energía nuclear encomienda a la JEN, entre otras misiones:

- El informe preceptivo al Ministerio de Industria en el trámite de las solicitudes formuladas por personas naturales o jurídicas, de Derecho público o privado, que se refieran a materias relacionadas con las aplicaciones pacíficas de la energía nuclear y el análisis de los riesgos y la seguridad intrínseca, así como la inspección en este aspecto de las instalaciones nucleares y radiactivas.
- El asesoramiento a las autoridades marítimas y aéreas del territorio nacional en la determinación de las zonas de puertos y aeropuertos donde podrán permanecer los buques o aeronaves nucleares, así como el prestar su colaboración a las citadas autoridades en la verificación de las garantías, comprobación de las protecciones y establecimiento de medidas de seguridad en puertos y aeropuertos.

Para el mejor cumplimiento de las misiones enumeradas, la JEN ha creado una Sección de Seguridad Nuclear, administrativamente dependiente de la Secretaría General Técnica.

La Sección de Seguridad Nuclear está constituida actualmente por el Grupo de Evaluación de Riesgos y el Grupo de Inspecciones, con un total de nueve técnicos superiores, auxiliados en materias concretas y específicas por los expertos de que dispone la JEN en su organización técnico-científica. Las principales actividades de la Sección son:

- Actividades encaminadas a garantizar que la ubicación, proyecto, construcción y explotación de las instalaciones nucleares y radiactivas se lleve a cabo sin riesgo indebido para la salud y seguridad de las personas y de sus bienes.
- Actividades en relación con el transporte sin riesgos de materiales radiactivos y combustibles nucleares, dentro del territorio nacional.
- Asesoramiento y colaboración con las autoridades marítimas en relación con la estancia en puertos nacionales del buque nuclear norteamericano SAVANNAH.

6. - Actividades en relación con las instalaciones nucleares

La ley referida sobre energía nuclear define como instalaciones nucleares:

- Los reactores nucleares.
- Las centrales nucleares.
- Las fábricas que utilicen combustibles nucleares para producir sustancias nucleares y las fábricas en que se proceda al tratamiento de sustancias nucleares, incluidas las instalaciones de regeneración de combustibles irradiados.
- Las instalaciones de almacenamiento de sustancias nucleares, excepto los lugares en que dichas sustancias se almacenen incidentalmente durante el transporte.

En la práctica las instalaciones nucleares españolas están sometidas a cuatro tipos de autorizaciones:

- AUTORIZACION PREVIA. Se concede en reconocimiento de la idoneidad del objetivo perseguido y del emplazamiento propuesto. Identifica los parámetros de diseño afectados por el emplazamiento y permite al solicitante adquirir los terrenos necesarios mediante las acciones legales apropiadas, incluida la expropiación forzosa por declaración de utilidad pública, permitiendo completar el estudio del emplazamiento.
- AUTORIZACION DE CONSTRUCCION. Faculta para realizar la construcción de la instalación de acuerdo con el proyecto aprobado y realizar las verificaciones prenucleares.

- PERMISO DE EXPLOTACION PROVISIONAL (PEP). Se concede en reconocimiento a que la construcción de la instalación es satisfactoria, para permitir, desde el punto de vista de la seguridad, la carga inicial del núcleo o la admisión de sustancias nucleares y la realización de las verificaciones nucleares, hasta alcanzar el pleno funcionamiento.
- PERMISO DE EXPLOTACION DEFINITIVO (PED). Se concede en reconocimiento de que la instalación cumple los requisitos impuestos y las especificaciones de funcionamiento, con garantía suficiente desde el punto de vista de la seguridad nuclear.

En cada una de las etapas la JEN ha de emitir un informe preceptivo de seguridad que ha de ser remitido a la Dirección General de Energía y Combustibles para que ésta resuelva sobre la autorización solicitada. Para la elaboración de los informes la JEN examina con detalle y evalúa la documentación técnica presentada en apoyo de la solicitud que ha sido dirigida al Ministerio de Industria. A su vez, la JEN mantiene contactos y reuniones de trabajo con el solicitante, pidiendo los datos complementarios que estime conveniente para su misión, visita el emplazamiento e inspecciona la instalación durante la construcción y la explotación. También puede recabar, de estimarlo oportuno, asesoramiento tanto de organismos nacionales especializados como extranjeros.

La solicitud de Autorización Previa debe estar acompañada de la documentación que sigue:

- Justificación de la instalación de acuerdo con las necesidades que trata de satisfacer; Memoria descriptiva de la instalación propuesta; Estudio económico y Características del emplazamiento seleccionado.

Desde el punto de vista de la seguridad nuclear, la JEN analiza el emplazamiento a la vista de sus características demográficas, hidrológicas, meteorológicas, sismológicas y geológicas, evaluando las posibles consecuencias radiológicas tanto en condiciones normales de funcionamiento como en caso de accidente. Como resultado, y a la vista de las manifestaciones habidas durante el período de información pública, la JEN emite un dictamen sobre la solicitud y sobre el emplazamiento estudiado, proponiendo, caso de ser favorable, los límites y condiciones que estime adecuados para su incorporación a la autorización, en la que se señala un plazo para presentar el proyecto completo.

La solicitud de Autorización de Construcción debe ir acompañada, entre otros, de los documentos preceptivos:

- Proyecto general de la instalación.
- Informe Preliminar de Seguridad.

Desde el punto de vista de la seguridad nuclear, la JEN analiza con detalle y evalúa el Informe Preliminar de Seguridad. Este informe debe contener datos precisos sobre las características del emplazamiento necesarias para completar y actualizar los presentados en la solicitud

de Autorización Previa; descripción de la instalación, incluyendo los criterios de diseño de los sistemas o componentes de los que depende la seguridad de la instalación; análisis de los accidentes previsibles y sus consecuencias; justificación de que la instalación no representa un riesgo debido para la población durante su funcionamiento normal; organización prevista para la explotación y adiestramiento del personal. Como en el caso anterior, la JEN emite un informe de seguridad sobre la solicitud, una vez analizados los documentos y recabada la información adicional que se estime necesaria del solicitante, con cuyos representantes se suelen mantener reuniones de trabajo. En el informe, caso de ser favorable, se fijan los límites y condiciones que deben ser incorporados a la autorización correspondiente.

Durante la construcción los inspectores de la JEN visitan periódicamente el emplazamiento para comprobar el cumplimiento de las especificaciones y demás condiciones del proyecto y presenciar la ejecución de pruebas prenucleares seleccionadas. Los inspectores también visitan las fábricas donde se manufacturan los distintos componentes de interés para la seguridad de la instalación. De cada intervención se levanta el correspondiente Acta.

La solicitud del Permiso de explotación provisional debe ir acompañada del Informe final de seguridad; también deben formular propuestas relativas a las especificaciones técnicas a seguir durante la explotación de la instalación, planificación de emergencias previsibles, reglamentos de explotación y normas de protección radiológica. La JEN revisa todos estos documentos y discute su contenido con el solicitante. Al mismo tiempo la JEN revisa los resultados de las verificaciones efectuadas en la instalación, en especial las correspondientes a las salvaguardias incorporadas, y lleva a cabo una inspección de la instalación. El solicitante ha de presentar también un programa detallado de la verificación nuclear prevista, que se evalúa y discute con el solicitante hasta alcanzar acuerdo mutuo.

Con antelación a la solicitud del Permiso de Explotación Provisional se solicitan las licencias de Supervisores y de Operadores de la instalación. Estas licencias las concede el Presidente de la JEN después de analizar los expedientes médico-psíquico y técnico de cada solicitante. En la mayoría de los casos los candidatos deben demostrar su preparación mediante exámenes escritos y orales.

Una vez que la JEN ha comprobado que la instalación se encuentra completa y se ha realizado la verificación prenuclear, que dispone del personal licenciado y que el contenido de los documentos es satisfactorio, emite su informe preceptivo al Ministerio de Industria proponiendo, si ha lugar, la concesión del Permiso de explotación provisional, pudiendo establecer límites y condiciones. Durante este período los inspectores de la JEN visitan con frecuencia el emplazamiento a fin de comprobar si sigue el programa de pruebas y verificaciones aprobado, y se cumplen las especificaciones de los documentos y los límites y condiciones establecidos.

Para obtener el Permiso de explotación definitivo el solicitante ha de demostrar que ha llevado a cabo la verificación nuclear de forma satisfactoria y completa. Debe también probar que el contenido de los documentos propuestos es adecuado o proponer las modificaciones que procedan. La JEN analiza de nuevo estos documentos y envía al emplazamiento sus inspectores a fin de realizar una comprobación final. Una vez satisfecha, la JEN emite su informe preceptivo al Ministerio de Industria sobre la concesión o no del permiso.

Durante la explotación normal los inspectores visitan la central periódicamente para comprobar si se sigue el contenido de los documentos aprobados y se cumplen los requisitos de la legislación aplicable.

Independientemente de las instalaciones nucleares de la JEN, existen actualmente en España las siguientes:

- Dos reactores tipo Argonaut, denominados ARGOS y ARBI, de 10 kWt, dedicados a la enseñanza e investigación desde 1962.
- Central Nuclear "José Cabrera", de 160 MWe con reactores de agua a presión. En funcionamiento desde 1968.
- Central Nuclear de "Santa María de Garoña", de 460 MWe, con reactor de agua en ebullición. Alcanzó la potencia de diseño en marzo de 1971.
- Central Nuclear de "Vandellós", de 480 MWe, con reactores de grafito-gas. Actualmente se encuentra en construcción avanzada.
- Han sido concedidas o están en trámites de Autorización Previa, cuatro solicitudes de centrales nucleares de potencia unitaria entre 1.000 y 2.000 MWe.

7. - Actividades en relación con las instalaciones radiactivas

La vigente Ley sobre Energía Nuclear define las instalaciones radiactivas:

- Las instalaciones de cualquier clase que contengan una fuente de radiación ionizante.
- Los aparatos productores de radiaciones ionizantes, con excepción de los aparatos de rayos X con fines médicos.
- Los locales, laboratorios, fábricas e instalaciones que produzcan, manipulen o almacenen materiales radiactivos.

Las instalaciones radiactivas quedan sometidas a un procedimiento de autorización administrativa similar al de las nucleares, aunque las autorizaciones se reducen y simplifican en función del riesgo potencial de la instalación. Las fábricas de producción de uranio, torio y sus compuestos, las fábricas de producción de combustible de uranio natural y

las instalaciones industriales de irradiación precisan las mismas autorizaciones que las instalaciones nucleares. Sin embargo, otras instalaciones que supongan procesos de construcción y montaje precisan autorización de construcción y de puesta en marcha y las que supongan un riesgo potencial menor precisan sólo la de puesta en marcha.

Actualmente existen en España unas 350 instalaciones; del orden de 300 han sido autorizadas con carácter provisional en la década 1960-1970. A partir del pasado año se han sometido al procedimiento administrativo referido unas 50 nuevas instalaciones, que están siendo inspeccionadas. A medida que las disponibilidades de personal técnico lo permitan se comenzará la revisión e inspección de las instalaciones que se pusieron en funcionamiento provisional entre 1960-1970.

8. - Actividades en relación con el transporte de materiales radiactivos y combustibles nucleares, dentro del territorio nacional

Las principales actividades se han referido al transporte de combustibles sin irradiar para las centrales de "José Cabrera" y "Santa María de Garoña", procedentes de EE.UU. y la República Federal Alemana, que han llegado a España por vía marítima y aérea. Actualmente se está efectuando desde Francia y por vía terrestre el transporte del combustible (uranio natural) para la central de "Vandellós". Con frecuencia tienen lugar transportes de material radiactivo con destino a las instalaciones radiactivas.

Los transportes de materiales radiactivos y de combustibles nucleares precisan, en cuanto tales, autorización administrativa del Ministerio de Industria, que es concedida a la vista del informe favorable de la JEN, cuando así procede. Puesto que no existe en España reglamento de transporte que regule la materia, se viene imponiendo que el transporte cumpla el "Reglamento para el transporte sin riesgos de materiales radiactivos del OIEA" (Edición 1967) y aquellos límites y condiciones que se consideren adecuados a cada transporte concreto. El cumplimiento de los requisitos impuestos es comprobado por los inspectores de la JEN, ya sea en el puerto o aeropuerto de entrada en el país, en destino o en ruta dentro del territorio nacional.

THE DESIGN OF NUCLEAR PLANT WITH INTEGRAL COOLANT CIRCUITS AS RELATED TO THE MAXIMUM CREDIBLE ACCIDENT APPROACH

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Abstract—Résumé—Аннотация—Resumen

THE DESIGN OF NUCLEAR PLANT WITH INTEGRAL COOLANT CIRCUITS AS RELATED TO THE MAXIMUM CREDIBLE ACCIDENT APPROACH.

Nuclear stations which are built to the integral design concept can have improved safety features as compared with non-integral designs. The safety status is a function of the safety criteria to be met and the credibility of any hazardous accidents postulated. In general throughout the world, the current safety criteria with regard to hazards to the public are based on defined limits to these hazards. By definition therefore, most designs are assessed in relation to a maximum credible accident (MCA) type of approach. The paper highlights the engineering philosophies and data which lead to the design of plant features which then quantify the MCA approach against limiting criteria. Faults are considered in two categories: those which can be considered highly improbable or incredible, due to basic design provisions and aims, and those remaining, which can be considered credible. In these arguments, the aspects of reliability of plant items and the probability of fault occurrences are considered. The logical development of potential accident sequences is explored and utilized in the analysis taking into account that credible occurrences could happen at any time. Thus, the implications of the MCA approach are illustrated and the necessary engineered features of the design are described. In summary, the integral design concept as defined can be engineered to meet the various interpretations and definitions of the MCA criteria which exist throughout the world.

ETUDE DES CENTRALES NUCLEAIRES A CIRCUITS DE REFROIDISSEMENT INTEGRES AU POINT DE VUE DE LA METHODE DE L' ACCIDENT MAXIMAL PREVISIBLE.

Les centrales nucléaires construites selon le concept d' étude intégrée peuvent jouir de mesures de sécurité améliorées par rapport aux centrales de construction non intégrée. La sécurité est une fonction des critères de sécurité qu' il faut satisfaire et de la vraisemblance des éventuels accidents dangereux admis. Dans le monde entier en général, les critères de sécurité actuels relatifs aux dangers pour le public se fondent sur des limites définies de ces dangers. Donc, par définition, la plupart des types de centrale sont évalués au point de vue de la méthode de l' accident maximal prévisible. Le mémoire fait ressortir les philosophies de construction et les données qui conduisent à l' étude de certains aspects de la centrale, lesquels servent à quantifier par la suite l' accident maximal prévisible par rapport à des critères limitatifs. Les défaillances examinées se classent dans deux catégories: celles qui sont considérées très improbables ou invraisemblables, par suite des dispositions et objectifs de base, et les autres défaillances, qui peuvent être considérées vraisemblables. L' auteur examine les aspects fiabilité des organes de la centrale ainsi que la probabilité de défaillances. Il étudie le développement logique des séquences d' accidents possible, et s' en sert dans l' analyse en tenant compte du fait que les défaillances vraisemblables peuvent se produire à n' importe quel moment. Il illustre ainsi les implications de la technique de l' accident maximal prévisible, et décrit les aspects nécessaires de construction incorporés dans l' étude. Pour résumer, le concept d' étude intégrée, tel qu' il est défini, peut être réalisé de manière à satisfaire les différentes interprétations et définitions des critères de l' accident maximal prévisible qui existent dans le monde entier.

ПРОЕКТ АТОМНОЙ ЭЛЕКТРОСТАНЦИИ С ПЕРВЫМ КОНТУРОМ, ЗАКЛЮЧЕННЫМ В КОРПУС РЕАКТОРА, В СВЯЗИ С РАССМОТРЕНИЕМ МАКСИМАЛЬНО ВОЗМОЖНОЙ АВАРИИ.

Атомные электростанции, которые строятся по проекту атомной электростанции с первым контуром, заключенным в корпус реактора, могут иметь лучшие характеристики безопасности по сравнению с реакторами других типов. Безопасность реактора зависит от выполнения принятых критериев безопасности и вероятности любых предполагаемых тяжелых аварий. В настоящее время практически во всех странах мира критерии безопаснос-

ти реактора для окружающего населения связаны с установлением определенных ограничений на степень опасности аварий. Поэтому большинство проектов основано на понятии о максимально возможной аварии (МВА). В докладе рассмотрены технические соображения и данные, которые позволили разработать проект электростанции с соответствующими характеристиками. Эти характеристики затем дали возможность количественно расчитать МВА по ограничивающим критериям. Все аварии на реакторе делятся на две категории: аварии, которые можно считать весьма маловероятными или невозможными (благодаря особенностям конструкции реактора или его назначению), и аварии, которые можно считать возможными. При рассмотрении этих вопросов учитываются как аспекты надежности узлов станции, так и вероятность возникновения аварий. Исследуется логическое развитие последствий потенциальной аварии, а результаты этого исследования используются в анализе данного случая, при этом предполагается, что авария может произойти в любой момент времени. Таким образом, иллюстрируется смысл понятия МВА и описываются необходимые инженерные характеристики данного проекта. Итак, концепция создания атомной электростанции с первым контуром, заключенным в корпус реактора, как она здесь определена, может пройти стадию инженерной разработки, чтобы удовлетворить требованиям различных толкований и определения критериев МВА, которые используются во многих странах мира.

EL DISEÑO DE UNA CENTRAL NUCLEAR CON CIRCUITO DE REFRIGERACION INTEGRADO Y SU RELACION CON EL CRITERIO DEL MAXIMO ACCIDENTE VEROSIMIL.

Las características de seguridad de las centrales nucleares con circuito de refrigeración integrado pueden ser mejores que las correspondientes a centrales no integradas. Las condiciones de seguridad son función de los criterios de seguridad previamente adoptados y de la verosimilitud de los accidentes postulados. Los criterios usuales de seguridad, en lo que al peligro para el público se refiere, se apoyan por lo general en todo el mundo en una definición previa de los límites de estos peligros. En consecuencia, la mayoría de los diseños se evalúan por definición, mediante el criterio del máximo accidente verosímil (MAV). En esta memoria se destacan los datos y supuestos técnicos que conducen al diseño de las características de la central que, a su vez, definen cuantitativamente el máximo accidente verosímil en función de los criterios limitativos. Los fallos se pueden encuadrar en dos categorías: los que se consideran muy improbables o inverosímiles como consecuencia de las especificaciones de diseño, y los restantes, que pueden considerarse verosímiles. En esta memoria se tratan ciertos aspectos de la fiabilidad de los componentes de la central y la probabilidad de que tengan lugar fallos. Se explora el desarrollo lógico de las secuencias de un accidente potencial, para usarlo a continuación en el análisis de seguridad, teniendo en cuenta que sucesos verosímiles pueden tener lugar en cualquier momento. Así pues se consideran las implicaciones del método del máximo accidente verosímil y se describen las características que, consecuentemente, deben incorporarse al diseño. En resumen, el concepto de diseño integral tal como se ha definido, puede llevarse a cabo de forma que se cumplan las distintas interpretaciones y definiciones del MAV que existen en el mundo.

1. INTRODUCTION

UK terminology defines a 'nuclear plant with integral coolant circuits' as one in which the reactor, coolant, circulation means and heat exchangers are within the primary reactor vessel (primary containment).

In defining the safety of a reactor and consequent hazard to the public adjacent to it, it is necessary to consider potential accidents as stemming from a whole sequence of events. In this sequence, the installed plant reacts to a defined fault occurrence to produce a calculable, even if negligible, level of activity release.

At the present time, most Licensing Authorities in the world define a site hazard restriction which must not be exceeded by any one credible accident (MCA). Sometimes, the concept of maximum hypothetical accident (MHA) is introduced in order to give a background against which to evaluate an MCA but an MHA is generally caused by a more improbable sequence of events following the same initiating fault or, alternatively, is based on a less credible (more severe) initiating fault.

With the integral gas-cooled reactor designs developed in the UK, engineering features on the prestressed concrete pressure vessel (PCPV) and its penetrations can be used to limit any

credible failures of the pressure envelope to relatively small identifiable cross-sectional areas. It can be shown that depressurisation accidents stemming from such failures constitute the most serious class of faults to be considered.

Similarly, by appropriate design of reactor protection, coolant and electrical supply components, it is possible to engineer the overall system to have a required degree of reliability. Hence it is possible to guarantee the magnitude and availability of post-accident cooling supplies to a designed reliability level.

By combining credible PCPV failures with various levels of post-accident cooling, it is then possible to obtain a spectrum of fuel damage and subsequent activity release corresponding to the differing courses of a depressurisation fault.

The purpose of this paper is to show that, by appropriate standards of engineering design on a Mk II Gas Cooled Reactor (AGR), the severity of a depressurisation fault can be limited to a required level of credibility and consequent activity release. Thus, if a maximum level of hazard to the public is specified by the Licensing Authorities, it is possible to associate this with a specific MCA and hence quantify the engineered safeguards built into the PCPV and the main coolant systems.

2. INTEGRAL COOLANT CIRCUIT DESIGN

2.1 PCPV design

The integral coolant design concept, as a feature of gas-cooled reactor technology, has developed primarily over the last decade.

The concept is closely associated with the adoption of prestressed concrete pressure vessels (PCPV) as the primary containment of gas-cooled reactors. PCPVs are currently being considered for water reactors.

A survey of concrete vessel technology is available [1]. The first gas-cooled integral design was for the Mk I Gas Cooled Reactor (Magnox) at Oldbury in the UK [1]. Subsequent UK designs have all been to this concept which now includes designs being built for the Mk II GCR and projected for the Mk III Gas Cooled Reactor (HTR), although the detail internal arrangements vary, depending upon the designer and the reactor type. Two current Mk II GCR stations now being built by The Nuclear Power Group Ltd. are Hinkley Point 'B' and Hunterston 'B'. Figure 1 is a cross-section of the Hinkley Point 'B' power station, showing pertinent features of the PCPV internal and external design including the major PCPV penetrations. A detailed exposition of Hinkley Point 'B' penetration design is given in [2] and further details of the reactor are given in [3].

The closure of large openings is usually accomplished either by a suitable design, whose complete failure can be claimed to be inconceivable, or by a more conventional steel closure with an additional independent device (secondary restraint) which would at least restrict the gas escape in the unlikely event of failure of the primary closure. This latter approach is adopted for the gas circulator penetrations.

Other, smaller penetrations, such as heat exchanger feed and steam connections or man access openings are designed in accordance with pressure vessel codes of practice and by combination with external monitoring and inspection, their failure also can be regarded as incredible. However, by design, even if a single weld fails, the resulting escape flow area is restricted.

Thus, it is a central premise that failure of any penetrations within the vessel envelope is incredible. Nevertheless, analysis of the depressurisation accident which would accompany such failures must be undertaken to evaluate MHA situations. The minimum gas discharge time-constant which occurs in the above classes of penetrations is about 240 seconds and applies to the boiler feed penetration. The use of this number in deriving an MHA is discussed further in Section 6 of this paper.

- 1 Reactor core
- 2 Supporting grid
- 3 Gas baffle
- 4 Circulator outlet gas duct
- 5 Boiler
- 6 Thermal insulation
- 7 Reheat steam penetrations
- 8 Main steam penetrations
- 9 Boiler feed penetrations
- 10 Cable stressing gallery
- 11 Gas circulators

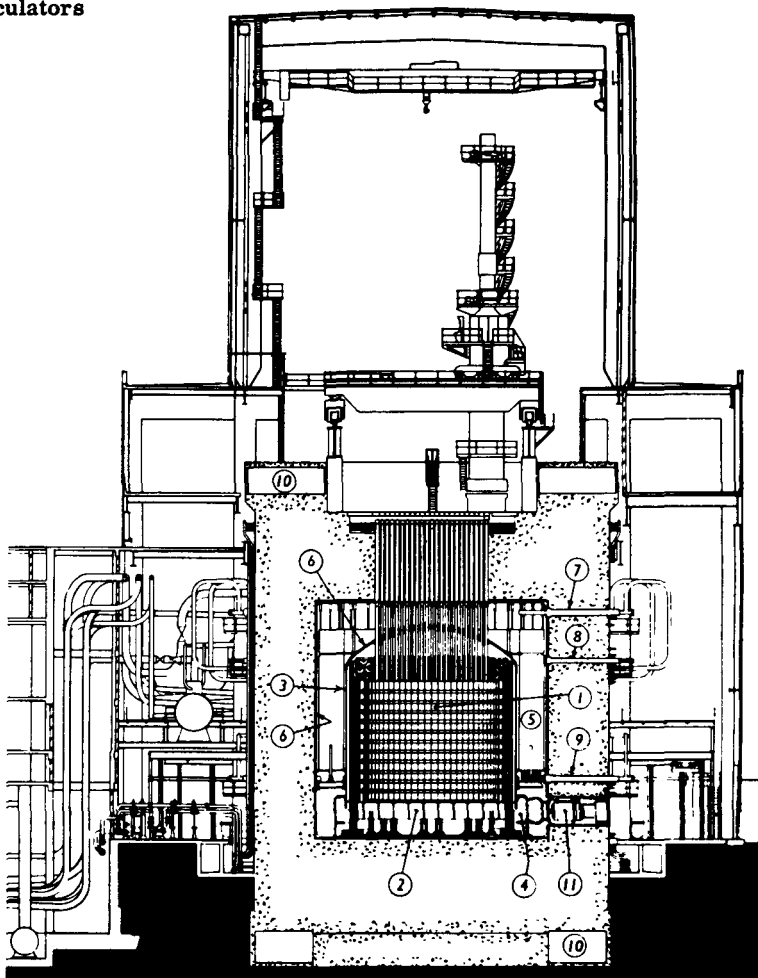


FIG. 1. Reactor elevation - Hinkley Point B AGR.

Since by design, the failure of any penetration within the vessel wall can be regarded as incredible, the remaining credible failures become those where the primary coolant circuit is connected to external plant via pipework outside the vessel. The most severe case for Hinkley Point Mk II GCR is the carbon dioxide bypass pipework.

However, it is argued that failure of protected and inspected plain pipework is incredible. The credible failure then finally becomes that of plant served by the pipework.

The diameters of this pipework are fixed by the design of the carbon dioxide bypass system plant requirements in terms of required flow and the bypass flow driving force (which in present designs is the circulator pressure rise).

The equivalent depressurisation time-constant, for complete fracture of this pipework, is some 600 seconds. Some depletion of core flow is occasioned, in addition, by virtue of the positioning of the delivery carbon dioxide pipework which draws carbon dioxide from the circulator outlet gas duct and hence depletes core channel flow.

The use of this number in deriving the MCA is discussed further in Section 5 of this paper.

2.2 Overall plant design

The overall plant includes, in addition to that described, the auxiliary systems.

In terms of safety, the relevant systems are those normally designated essential services, engineered safeguard systems etc. (the definitions are not necessarily consistent, being respectively UK and USA orientated).

However, whichever terminology is used, there are three main categories of these for gas-cooled reactor systems:

- the reactor protection system
- the primary coolant system (gas)
- the secondary coolant system (feedwater)

The safety analysis defines the design requirement for these systems in relation to the safety criteria.

With systems such as those above, it is possible to determine relative reliability and action-on-demand probability figures. These can then be used in relation to the probability of known fault occurrences to quantify credible accident situations.

In this context it is interesting to note that the on-demand reliabilities for emergency feed systems could be required to be higher for pressurised coolant rather than the depressurising fault situations. That is, the requirement would be dictated in the first instance by economic rather than public safety grounds.

The detailed design of the feed and coolant systems used after shutdown or in emergency is also a function of the station electrical system.

2.3 Fault definitions

The previous sections have indicated that there are faults which can be, or need to be, eliminated by design. These faults are failures of the PCPV itself, all penetrations, and complete loss for an indefinite period of essential systems such as electrical supplies, feedwater and reactor coolant.

The credible faults relevant to safety analysis generally fall into the categories of reactivity, loss of coolant flow, loss of feed flow (vessel pressurised) and loss of coolant flow primarily (vessel depressurising).

3. THE MAXIMUM CREDIBLE ACCIDENT APPROACH

In this approach, there are defined doses, body and thyroid, adult or child, which must not be exceeded by any one accident.

In principle this means that an MCA must be defined.

In certain countries higher dose levels are acceptable for defined hypothetical accidents (MHA).

A typical limitation is that members of the public should not be exposed to a whole body dose exceeding 15 rem and that total thyroid dose (child) should not exceed 25 rem. These typical limits may be used in relation to the MCA or perhaps, in certain circumstances, also to the MHA. Taking such a criterion, the dose is a function of distance from the site boundary which can be as close as about 60 m in the UK.

More important, it is a function of total activity released and an MCA (or MHA) analysis must generally seek not to exceed the total permitted activity release derived from such design limitations. In practice the thyroid dose limit is generally the more critical.

4. SAFETY ANALYSIS TO DERIVE AN MCA

4.1 Accident analysis logic

To meet an acceptable activity release, the analysis of accident sequences must be carried out in a systematic manner, taking account of options at each stage and leading up to a definition of MCA or MHA.

The following logic can be applied:

- (i) There are faults which can be argued to be eliminated by engineered features and/or practices.
- (ii) Engineered safeguards can be defined and used appropriately in the fault analysis.
For example in a Mk II GCR the system of emergency forced cooling provisions consists of:
 - (a) the eight main gas circulators supplied in emergency by two or more diesels or gas turbines. (i.e. if station supplies are lost).
 - (b) the boiler emergency feed pumps supplied in emergency by the above prime movers or separate ones. (details suit individual designs and depend upon whether the design is for single station or not).
- (iii) Assumed mal-operation in the fault sequence can be defined in qualitative fashion or in terms of active component failures. These arguments might be applied to:
 - (a) on-site generation
 - (b) any system or engineered safeguard relevant to the accident

A single active component can be defined as 'a functional unit of a system' (for example a boiler quadrant).
- (iv) Any MCA can have an associated MHA rating if engineered safeguard(s) or equivalent 'systems' are invoked to keep the station safe in the MCA case. Alternatively, an MHA could be derived for an incredible initiating fault.

The MHA is 'hypothetical' and, as such, is judged not to happen. The condition which transforms the MCA into an MHA is associated with partial or complete failure of any system or any engineered safeguard which is invoked in the MCA. This failure is argued to be mechanistic in the sense that failure of part of a system results in a consequent defined action.

4.2 Credible accident survey

Faults are classified essentially in relation to the possibility of fuel element failure and of the escape of fission products to atmosphere. Under these conditions, the most significant isotope which may affect public health is Iodine 131. For the Mk II GCR long term failure mechanisms of the fuel can for pressurised conditions are associated with grain enlargement and loss of ductility, whilst those for depressurising conditions are associated with creep failure of the cans under internal pressure. For each of these conditions, design data relating can temperature, acceptable time and stress (the latter relevant to the depressurising case) are used to relate can integrity to the transient temperature changes.

Fuel can melting would occur at about 1390°C but, for safety purposes 1350°C is used as a short-term limit. The protective and trip systems aim to ensure that no can temperature exceeds 1100°C. The design steady state temperature for a 'peak random' can at 100% power is 825°C. Calculations are necessary for each assumed fault to relate the transient temperature performance to the design limits. Out of a total of 90,000 pins in the reactor approximately 1 in 1000 are designated as 'peak random'. A pin is defined as the stainless steel can which contains the fuel pellets of UO₂ together with the end caps.

Calculations done for symmetric and asymmetric reactivity faults and all circulator failure faults including inlet guide vane (IGV) faults, generally indicate that even when detected and tripped by a second or third line of protection no damage to fuel occurs. This is a direct consequence of the aim to provide protection such that no fuel can temperature exceeds about 1100°C during the transient, this level thus effectively being an economic limit.

Loss of coolant flow faults for pressurised conditions in a single channel, such as would be caused by channel blockage or inadvertent closure of a channel gag, could give rise to varying degrees of fuel can damage. However, even if a whole fuel channel fission product inventory was released (10⁵ curies of Iodine 131) the hazard arising would be acceptable in view of the primary containment barrier provided by the PCPV and the provision of a suitable venting stack to atmosphere for subsequent controlled release of noble gases.

Significant bypassing of core flow is, by design, incredible since special attention is paid to construction materials and inspection techniques employed for the gas baffle, gas ducts, boiler casings and gas seals to ensure that failures giving rise to excessive fuel temperatures and hazardous situations do not arise. Similarly, it is claimed that hazardous situations cannot occur due to failure of the support and restraint structures or of the boiler support structure. However, this type of hypothetical fault has been analysed and it can be shown that even for very large bypassing of the core, the severity of the excursion would be acceptable without making special demands upon the effectiveness of fault detection and subsequent trip actions.

To ensure the viability of the gas baffle, gas ducts and boiler during the station lifetime care needs to be taken in ensuring that operational transients are satisfactory. In particular, shutdown temperature transients, notably those caused by feed failure, are significant in that boiler outlet/reactor inlet gas temperature transients can give rise to critical component temperature differentials. In turn, shutdown regimes can be affected by the design and operation of the essential cooling systems of the reactor. Thus in summary, such systems need to be designed to a degree of reliability which does not hazard vessel components. Thus again, economic factors have a bearing on a plant design such that, if they are met, the safety risk is minimised.

However, to summarise the above discussion, it can be shown that reactivity and loss of coolant flow (vessel pressurised) faults do not cause environmental hazards which are serious enough to make them the MCA. The most serious faults are those arising from loss of coolant flow (vessel depressurising) and the MCA and MHA definitions generally arise from variants of this type. Further discussion in this paper is therefore restricted to this class of accidents, whilst noting that there are other classes of fault which can influence the reactor design in their own right.

10 MIN TC, 2% INITIAL DEPLETION INCREASED
TO 4% AT CIRCULATOR TRIP

T_1 RAMP OF $100^\circ\text{C}/600$ SECS; THEN CONSTANT LEVEL

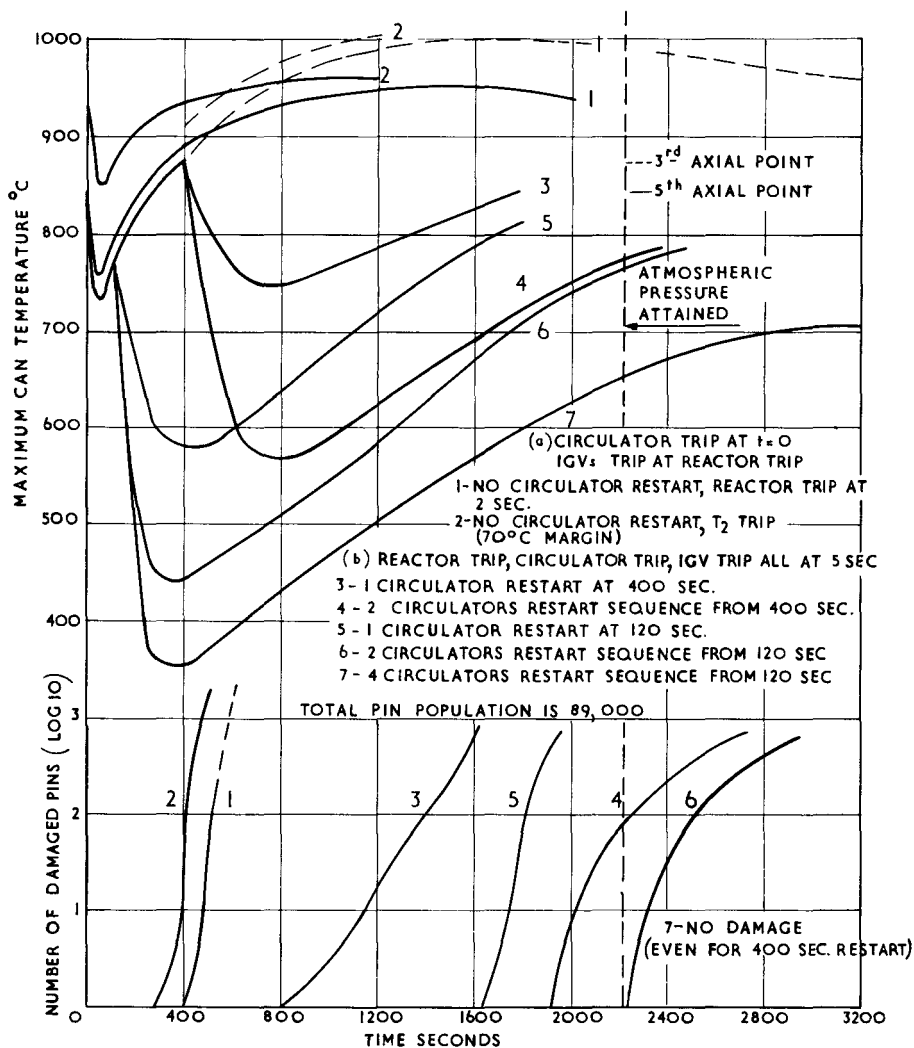


FIG. 2. PCPV depressurisation, worst credible failure.

The analysis of this class of credible faults is therefore directed to quantifying the number of fuel failures that occur during and after the course of the transient. In taking account of the various options outlined in 4.1 above, a spectrum of conditions and ultimate activity releases can usually be found as illustrated by the following section.

5. MAXIMUM CREDIBLE ACCIDENT ANALYSIS

The only credible depressurisation failure which may involve fuel can failure is discussed in Section 2.1 and can be shown to be failure of the carbon dioxide bypass pipework in the bypass plant room. This is therefore the basic accident and leads to vessel depressurisation with a time constant of about 600 seconds.

Inherent in the accident definition, however, are such factors as the following:

- (i) Any coincidental fault e.g. loss of supplies to all circulators.
- (ii) The reactor tripping mode e.g. by rate of change of coolant pressure (Dp), channel gas outlet temperature detection (T_2), or loss of circulators. Circulators and feed are normally tripped on reactor trip.
- (iii) The extent and time of re-establishment of coolant flow (Boiler feed re-establishment should also occur). This coolant flow re-establishment is via a.c. supplies or via power from at least one out of two prime movers (gas turbine or diesels). In this latter event, the re-establishment is therefore via an engineered safeguard system.

Four circulators out of eight should normally start (one per quadrant) following any reactor trip and subsequent re-establishment of feed

5.1 Cases considered

Considerable analysis has been carried out for the specified depressurisation accident combined with various permutations of the above and the following examples are presented:-

(Analyses have been carried out on the basis of a nominal peak rated channel whose can temperatures are about 100°C less than the peak random temperature. Run numbers used below refer to Fig. 2.)

- (i) Coincidental depressurisation plus loss of circulators with reactor tripping at:
2 seconds by -20% speed change — run 1
7 seconds as tripped by T_2 — run 2
(this latter is a third line protection case for illustrative purposes only)
- (ii) Depressurisation only with reactor tripping after a further five seconds via Dp trip — runs 3–7. (Trip by T_2 could occur at about 180 seconds, the subsequent temperature transient being similar)
- (iii) Diesels or gas turbines started and available to restart circulators within two minutes after depressurisation initiation. However, allowing for the depressurisation detection, the coolant mass flow patterns have been defined according to:

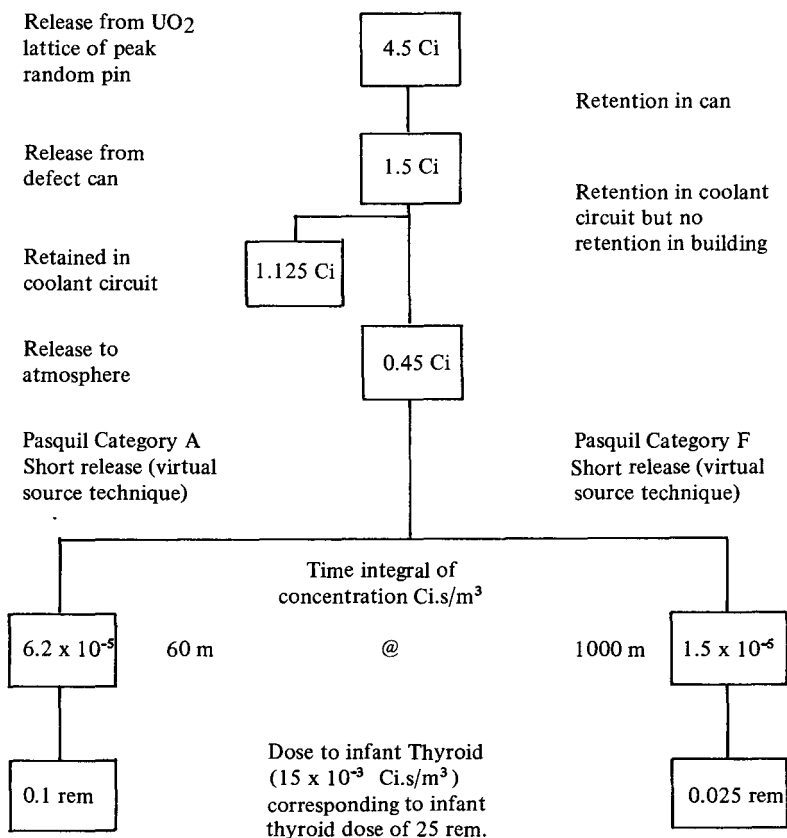
Runs 3–4 low pressure level detection and hence delayed circulator restart (at a pessimistic 400 seconds: compared with a design value of 300 seconds)

Runs 5–7 Dp detection and hence earlier initiation at 120 seconds.

Transient temperature results together with estimates of the number of damaged pins are plotted in Fig. 2. The majority of the temperature curves are plotted for the '5th axial' point approximately $3/4$ of channel height from inlet; for runs 1 and 2 the maximum temperature moves lower down the channel and temperatures at the '3rd axial' point (approximately $5/12$ of channel height) are also plotted. The number of damaged pins is derived by integration of their temperatures and can hoop stresses with time combined with the experimentally measured probabilities of pin failure for such integrals.

TABLE I
DEPRESSURISATION FAULT

IODINE 131 DOSE FROM A SINGLE PEAK RANDOM PIN FAILING
BEFORE ATMOSPHERIC PRESSURE IS REACHED



NB: 1) Height of release from building is not significant.

2) Release from UO₂ lattice for an average pin at peak axial rating is 0.6 curies.

5.2 MCA assessment

Table I shows the derivation of Iodine 131 dose rates to be expected at the site boundary (taken as either 60 m or 1000 m) as the result of a single peak random pin failure during a depressurisation accident. At a 60 m site boundary this is about 0.1 rem.

Using the dose rates set out in Table I, and a permissible total release limit of 25 rem, an acceptable MCA should not lead to failure of more than 250 pins. However, it is desirable to work to lower values in order to provide a large tolerance band to cover any limitation of damage calculational techniques or estimates of Iodine 131 release.

From inspection of Fig.2 it can then be seen that Run 7 leads to an acceptable MCA which would provide very large safety margins both during and after the transient. Run 7 corresponds to 4 circulators being restarted after 120 seconds and even a delay to 400 seconds does not markedly alter the situation.

Taking a 'single component failure' criterion as applying to whole boiler/circulator quadrants, this MCA therefore only requires two quadrants to be effective. Thus, even in the unlikely event that the immediate pre-fault situation was only three quadrants operative, there is still a margin for a plant failure to occur such that one quadrant is inoperative.

Even greater circulator restart times than 400 secs could be accepted, subject to the amount of damage built-up before they start. The MCA definition of four circulators starting within 400 secs covers the range of initiating faults and trip sequences postulated in 5.1.

6. MAXIMUM HYPOTHETICAL ACCIDENT ANALYSIS

Two variants of this might be distinguished, as follows:

6.1 Failure of CO₂ bypass pipework

The initiating event for this case is the same as for the MCA discussed above but with more severe subsequent plant failures leading to either:

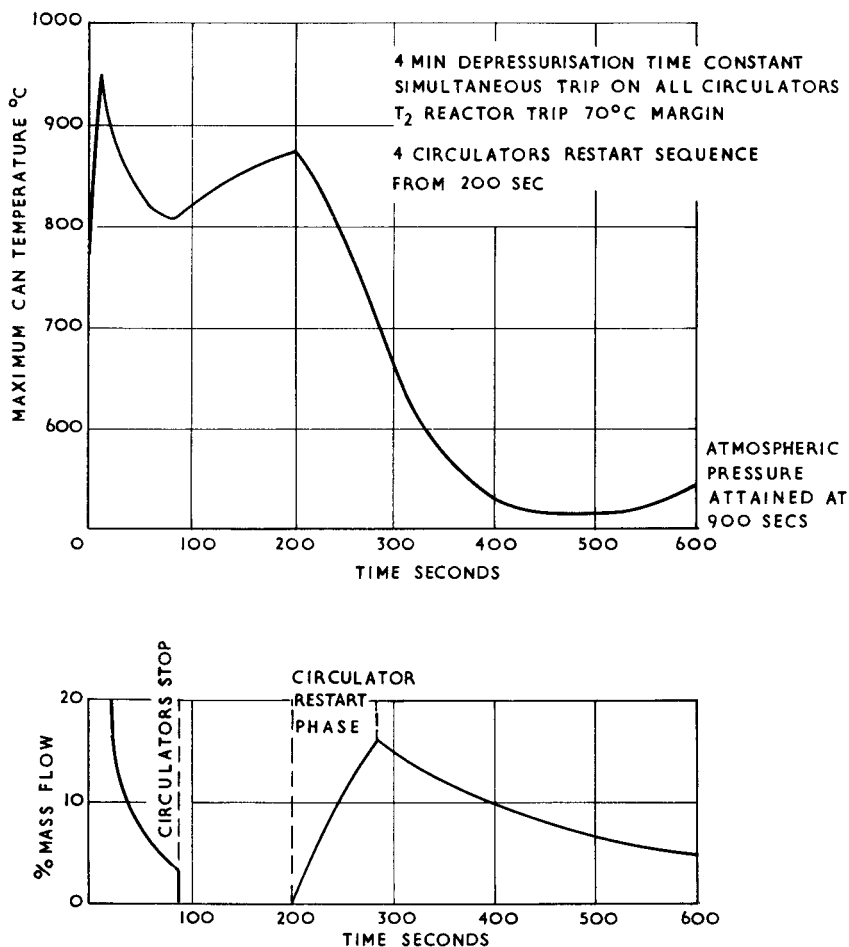
- (i) one quadrant with two circulators
- (ii) or two quadrants with one circulator per quadrant

Two circulator restart cases are shown by runs 4 and 6 in Fig. 2. Both runs give rise to increasing fuel damage in the short and long term, irrespective of effective mass-flow re-establishment times.

For Run 4 (circulator restart delayed for 400 seconds) the number of pins damaged before atmospheric pressure is reached would be approaching 100. In general terms, this would give less than 200 curies Iodine 131 release from the UO₂ lattices with a consequent short-term site hazard of less than the limit postulated for the MCA (being less than 5 rems for the 60 m site boundary case).

In the long-term however, beyond about 2500 seconds, both run 4 and run 6 show further pin failures. Whilst fuel temperatures continue to increase this will result in further release to the atmosphere. Thus there will be a long-term additional risk unless satisfactory site measures are taken; e.g to restart manually additional gas circulators or to seal the breach. This latter approach could then lead in the ultimate to plant design provisions such as secondary containment.

Even more extreme MHA situations are represented by cases where only one circulator restarts (runs 3 and 5). With such an extreme pessimism, however, this case is unacceptable even before atmospheric pressure is reached with delayed re-establishment of coolant flow (i.e. waiting for depressurisation signal until effectively 400 seconds, since extensive damage in excess of thousands of pins could arise).



NB. THERE ARE NO PINS DAMAGED IN THIS
CASE BY INTENTION (CF. RUN 2 OF FIG.2)

FIG. 3. PCPV depressurisation, improbable failure.

6.2 Failure of feed penetration

As explained in Section 2.1 this failure is regarded as incredible, but if it were to occur, would give a maximum depressurisation time constant of about 240 seconds.

An example of the accident is given in Fig. 3 for the following conditions:

- (i) Coincidental depressurisation plus loss of circulators.
- (ii) Reactor trip by T₂ (a third line protection).
- (iii) Four out of eight circulators started, as for the MCA, at a pessimistic 200 secs. (c.f. a design value, for this depressurisation rate, of 120 secs).

This transient does not result in pin failure despite the more rapid depressurisation and it therefore can be seen that further plant failures in the essential systems need to be invoked before substantial fuel can damage would occur in the short term and long term. See Fig. 2 for a general appreciation of this noting that the early part of the temperature transient in Fig. 3 is similar to that of run 2 on Fig. 2 and hence fuel can damage build up would be worse due to the faster depressurisation.

7. ESSENTIAL SYSTEM RELIABILITY

The three essential systems are each designed to a suitable standard of reliability.

The failure to trip the reactor is rendered incredible by designing the reactor protection systems to have an on-demand-failure-rate of about 1 in 10^7 . This, by judgment, is deemed adequately to cover the safety risks allied to frequent faults (e.g. spurious trips which could be of the order $10/y$).

Section 5 derives an MCA on the basis that four out of eight circulators will restart within about 400 seconds after the credible depressurisation (via CO_2 bypass pipework) has commenced.

It is thus necessary to show that the operation of the essential safeguard systems for emergency coolant flow and feed flow matches this to the required level of credibility. The design of these systems is required to be of a sufficiently high standard such that there is adequate reliability. The order of this reliability is subject to various ground rules and plant operator action. The on-demand failure rates for the two systems including emergency electric supplies generally are each of the order 1 in 10^5 , for a stated minimum operational requirement of the systems (e.g. four circulators in two quadrants for the MCA case).

Taken in conjunction with the assumption that the depressurisation accident is postulated, itself an unlikely event, to be coincident with loss of normal electric supplies (also a low probability), it may be judged that failure of the essential safeguard systems is highly incredible especially for MCA and MHA criteria.

The selection of a level of incredibility of plant operation is often a matter of judgment and it could be argued that the particular provisions, which are made in the essential safeguard systems are insufficient to meet the level demanded by potential MCA and MHA definitions.

This objective may be met by introducing more redundancy into the systems or calling upon more operator action (given sufficient time). Alternatively the scale of the MCA can be cut down by the use of various back-up provisions such as:

- (i) Secondary closures as distinct from secondary restraints on relevant penetrations, thus eliminating coolant depressurisation with respect to such penetrations.
- (ii) Special design of such systems as the necessary coolant bypass circuitry, to minimise potential accident risk. This might include:
 - pipe diversification
 - different positioning to eliminate 'depletion'
 - provision of auto-closing flap valves on relevant pipework such as the carbon dioxide bypass
- (iii) Provision of post-accident sealing of pressure vessel and use of vessel purging and blow-down through iodine filters. This latter can be related most easily to particular potential faults rather than to all faults, since otherwise, secondary containment would, in the ultimate, become necessary.

In any event it is always possible to define the plant features and the course of an MCA such that it is matched to a given design limit for activity release.

8. CONCLUSION

The interrelationship of an integral coolant circuit design and the MCA safety approach has been shown to be as follows:

- (i) the design of the PCPV and its penetrations minimise the possibility of credible failures which would have a direct bearing on the radiation risk to the public in the vicinity.
- (ii) the essential safeguard systems or engineered safety features with appropriate diversity are:

Reactor protection systems capable of ensuring reactor shutdown for all postulated fault conditions.

The use of the normally running and monitored main gas circulators for emergency cooling; power supplies being provided from diesels or gas turbines.

Emergency heat removal systems which use the main heat exchanger facilities with feed pumps supplied by the above prime movers or separate ones.

These systems are individually designed to a suitably high standard of reliability.

- (iii) Various back-up provisions can be readily built into a PCPV design such that more pessimistic fault situations can be defined or more safety margins can be achieved.
- (iv) In essence, MCA sequences are very unlikely events, which however, for the purpose of analysis, are deemed to happen. Having been deemed to happen, the reliability of the engineered safeguard systems must be of a high order to show that the specified sequences of plant action are realistic and can, in fact, occur. Their probabilities of failure will generally be much less than would be the case if full credit were being taken for the very remote probability of the initiating fault occurring in the first place.

Thus a definitive rather than a statistical environmental risk target is postulated. The MCA approach therefore can demand a higher degree of reliability which improves the safety of the plant albeit perhaps at some expense.

9. ACKNOWLEDGEMENTS

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CONTAINMENT AND SAFETY IN SWEDISH NUCLEAR POWER PLANTS

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Abstract—Résumé—Аннотация—Resumen

CONTAINMENT AND SAFETY IN SWEDISH NUCLEAR POWER PLANTS.

In accordance with the wishes of the Swedish Reactor Safety Committee, ASEA-ATOM follows the USAEC basic rules for nuclear safety. A review is given in the paper of the opportunities and difficulties encountered when utilizing in this way the standards of a foreign organization to which access for background research is not granted.

In the area of normal operation the Swedish policy on release of radioactive matter from nuclear power plants is developing somewhat differently from that in some other countries. Where the current trend is to lower the permissible exposures of the nearest man, Swedish rules will probably concentrate on the lowering of population exposures. The paper reviews briefly what this means to the nuclear plant designer.

One contribution of ASEA-ATOM to nuclear safety which is described in the paper is in the field of reactor containment vessel integrity. The background of the imbedded liner concept and other features of the concrete pressure-suppression containment vessel are reviewed. The basic philosophy of the Värtan underground plant design is also presented.

ENCEINTE DE CONFINEMENT ET SURETE DES CENTRALES NUCLEAIRES SUEDOISES.

ASEA-ATOM est tenu d'observer les règles fondamentales de l'USAEC pour la sûreté nucléaire, sur la requête du Comité suédois de sûreté des réacteurs. Le mémoire rend compte des possibilités et difficultés rencontrées en utilisant de cette façon les normes d'un organisme étranger dont l'accès pour révision n'est pas autorisé.

Dans le domaine des opérations normales d'évacuation des matières radioactives des centrales nucléaires, la politique suédoise se développe d'une façon un peu différente de celle de certains autres pays. Tandis que la pratique courante est de réduire l'exposition aux rayonnements de l'homme le plus proche, les règles suédoises se concentreront probablement sur la réduction de l'exposition de la population. Le mémoire en expose brièvement les conséquences pour les constructeurs de centrales nucléaires.

Une contribution d'ASEA-ATOM à la sûreté nucléaire est décrite dans le mémoire; elle a trait à l'intégrité de l'enceinte de confinement. Les raisons de la conception du chemisage intérieur et des autres caractéristiques de la cuve en béton sont exposées. La philosophie à la base de la conception du projet de la centrale souterraine de Värtan est également présentée.

ЗАЩИТНЫЕ ОБОЛОЧКИ И ВОПРОСЫ БЕЗОПАСНОСТИ НА ШВЕДСКИХ АТОМНЫХ ЭЛЕКТРОСТАНЦИЯХ.

В соответствии с пожеланиями Шведского комитета по безопасности реакторов объединение ASEA-ATOM руководствуется основными правилами по ядерной безопасности, разработанными КАЭ США. В докладе рассматриваются возможности и трудности, встретившиеся при работе с данными стандартами зарубежной организации, к фундаментальным исследованиям которой доступ закрыт.

В области нормальной эксплуатации станций шведская политика по вопросам сброса радиоактивных отходов атомных электростанций несколько отличается от той, которая приводится в других странах. В то время как наблюдается современная тенденция к снижению допустимых доз облучения для работающего персонала, в шведских правилах сосредоточивается внимание на снижении доз облучения для населения. В докладе кратко рассмотрены эти вопросы и их значение для проектировщиков атомных электростанций.

Одним из вкладов объединения ASEA-ATOM в обеспечение ядерной безопасности, который описан в докладе, является разработка вопроса целостности защитной оболочки реактора. Рассмотрены концепция внутренней герметизирующей оболочки и другие особенности корпусов из предварительно напряженного бетона. Представлены основные аспекты проекта Вертанской подземной атомной электростанции.

CONTENCION Y SEGURIDAD EN LAS CENTRALES NUCLEARES SUECAS.

La Comisión de seguridad de reactores de Suecia ha pedido a ASEA-ATOM que siga las normas fundamentales de la seguridad nuclear establecidas por la Comisión de Energía Atómica de los Estados Unidos de América. La memoria analiza las posibilidades y dificultades que se encuentran al utilizar normas de un organismo extranjero, al cual no existe acceso autorizado para cuestiones de revisión.

En el campo de la explotación normal y de la evacuación de materiales radiactivos producidos en las centrales nucleares, la práctica sueca se desarrolla de forma un tanto distinta a la de otros países. Mientras que la tendencia actual se encamina hacia la reducción de la dosis tolerada con relación al individuo más cercano, las normas suecas se concentran preferentemente en la reducción de las dosis a la población. La memoria analiza brevemente lo que ello significa para el proyectista de centrales nucleares.

En la memoria se describe también una contribución de ASEA-ATOM a la seguridad nuclear en el campo de la integridad de las vasijas de contención. Se revisan los fundamentos del concepto del recubrimiento incorporado y otras características de la vasija de hormigón que se utiliza para la contención por el sistema de relajación de la presión. También se presentan los principios seguidos en el proyecto de la central bajo tierra de Värtan.

1. REACTOR CONTAINMENT VESSELS

1.1. General background

A reactor containment vessel is designed to withstand the stresses caused by primary system ruptures that are considered credible. It must also be tight when subjected to the accident stresses and temperatures, i. e. tight enough so that not more than a certain part of its internal atmosphere leaks to the vessel environment, as the containment vessel cooling systems bring down the vessel pressure.

The permissible leakage depends on the plant location and on external arrangements such as secondary containment and filter arrangements. It usually lies somewhere in the range of 0.1 - 10% per 24 hours at the design pressure of the vessel.

No realistic test can be made of a containment vessel under actual accident conditions. The measures taken to verify its integrity will therefore have to take other forms, which include:

- (1) a cold pressure test;
- (2) recurrent cold tightness tests;
- (3) calculations of the combined thermal and mechanical stresses in the vessel after an accident;
- (4) protection of the vessel against missiles, shock waves and pipe forces generated by an accident;
- (5) protection of the vessel against an overheated core which penetrates the reactor vessel.

Of these five, only the cold tightness test gives a quantitative figure as a result. This is probably why this test has drawn most of the attention in containment vessel tightness specifications. There are severe limitations in the value of such cold tightness tests, however. If missiles, etc., inflict any damage at all to the vessel, they will cause leak rates which are almost certainly orders of magnitude greater than the figures quoted above as acceptable. This weakness of the quantitative test is only partially reflected in the suggested standards for recurrent tightness tests for reactor containment vessels.

1.2. Non-urban sites

For non-urban sites, the criterion of total severance of the largest primary system pipe applies in most countries. This criterion then decides the requirements for the containment vessel integrity.

For such sites, ASEA-ATOM has selected a prestressed concrete containment vessel with a pressure suppression pool in the vessel bottom and an imbedded steel liner. One way in which this liner may be placed in the concrete has been described elsewhere [1]. It suffices here to say that the cables are stressed when the grouting around the liner is hardening and that liner and concrete are solidly compact with no open spaces along the liner surfaces. It should also be mentioned that the imbedded plate liner is welded to the liner pipes of the vessel wall penetration holes.

The only containment vessel actually built by ASEA-ATOM is the one for the Oskarshamn I plant. ASEA-ATOM has also given specifications for the vessels of the plants Ringhals I, Oskarshamn II and Barsebäck I, however, and these containment vessels are all modelled after the Oskarshamn I design.

The concrete vessel is designed to withstand the pressure generated by the rupture of the largest primary system pipe following applicable standards for prestressed concrete structures. A certain degree of metal-steam reaction is allowed for in the design pressure. An overpressure which exceeds the design pressure by 50% will still leave the liner intact; overpressure 100% higher than the design value will give increased leakage but the vessel will not disintegrate.

One important advantage of the imbedded liner concept is the protection which the tight membrane is given by the concrete; 20 - 25 cm of the wall concrete is inside the liner and absorbs impacts from missiles and from high-temperature jets of steam and water. The massive walls make the vessel less sensitive to shock waves than thin-walled vessels. The imbedding also protects against corrosion attack.

One drawback of a concrete vessel turns out to be the poor respect with which it is treated by people. It is regarded as any building and there is always a risk that somebody might drill a hole through the tight membrane. Leaks can be tightened by epoxy injection, however, and the vessel is no more difficult to mend than are parts of most steel containment vessels.

The deep pool forming under the reactor after accidents and a concrete floor with an imbedded tight liner under the pool should also be the best possible insurance against a Chinese syndrome type of melt-through.

The vessel wall pipe penetrations consist of conical sleeves welded to the pipe and to the wall penetration liner pipe. The liner pipe is anchored to the wall with irons imbedded in the concrete. No extra anchoring of pipes or of isolation valves to the building are necessary. The penetrations are made strong enough to withstand all forces, moments and torques which the pipes can inflict.

1.3. Urban reactor sites

For urban reactor sites, the kind of accident that should be assumed for design basis has not been settled, at least not as universally as the pipe rupture has been accepted for non-urban sites.

If the pipe rupture criterion is finally accepted for urban sites, the principles for the design of containment vessels should be the same for these sites as for non-urban sites. If, on the other hand, a criterion such as a pressure vessel rupture were to be enforced for urban sites (or a sabotage criterion or possibly one of a very severe earthquake), the containment vessel designs would change appreciably. There have been considerable efforts to do things the other way around and to adapt criteria to different containment vessels and reactor vessels.

If a vessel failure is assumed to be slow or not to cause a very large fracture opening, the changes in containment vessel design may be small. The pressure suppression schemes of different kinds with water and ice may not be applicable, however, and the large dry containment vessel may be the only solution.

If it is assumed that any kind of vessel failure can happen, the repercussions will be quite different. The missile forces and steam shock waves will be very difficult to cope with, and the plant design will be strongly influenced by the new criterion. Protection against steam shock wave reflections may thus be difficult to reconcile with radiation shielding requirements and missile protection requirements with the need for thermal insulation and possibly with external pressure vessel inspection. It may even be that common building techniques are not sufficient to guarantee containment vessel integrity.

Large vessel ruptures of this kind could be spontaneous or they could be caused by large transport accidents such as an airplane crash and, possibly, by reactor excursions.

In most of Scandinavia, these difficulties can be circumvented by going underground into a rock chamber. Granite bedrock has in most places only a thin cover of earth and in many essentially none. Rock chambers have long been used for certain factories, defense installations and civil defense shelters. Most hydroelectric power plants and one 800-MW, oil-fired power plant are located in rock chambers. In the construction of rock chambers, conventional technology is used, and this technology has shock wave implications in the defense applications.

A rock chamber can fulfil certain functions more easily than other containment vessels for the following reasons:

- (1) The large mass of rock cavity walls makes a rock containment vessel strong enough to withstand even impacts from large sabotage explosions and from ruptured reactor and steam generator vessels. The physical integrity of a containment vessel can thus be safeguarded even for these kinds of primary accident.
- (2) A rock chamber containment vessel can be made to leak inwards in accident situations by having a groundwater pressure in the rock outside the vessel. This pressure can be made higher than the pressure in the containment vessel during the critical period after an accident. This feature is achieved by grouting the rock around the cavity to make it fairly tight and by ensuring that there is water in the cracks in the rock outside the grouted zone.

The weight of the rock above must be sufficient to prevent the cavity rock cover from being lifted up by the pressure in the chamber. Rock chambers have been tested to very high pressures.

The "negative" leakage containment vessel principle is shown in a figure in a previous paper [2]. An explosion or a vessel missile could chip away

part of the grouted zone rock, but the grouting can be made so deep that most of it will still be left.

Pre-grouting of rock is today a well-known and conventional technology. It is performed before blasting tunnels, etc., for several reasons, such as keeping the groundwater level unchanged so as not to damage surface buildings. It also makes the rock surface stronger and more even.

The tightness of the grouted zone can meet a containment vessel criterion of admitting only a few hundred m³ of water per day from the surrounding aquiferous rock. A higher leak rate could make the water difficult to handle after an accident.

If there is not enough water in the rock, pools and water injection could be utilized to provide water. Rock which is originally aquiferous is preferable, however, as wetting a rock mass is a very lengthy process.

Even when the rock cannot be grouted, there is still an advantage in rock chambers as compared with surface buildings. This advantage originates in the fact that not only can the rock chamber be made to withstand the momenta from large missiles, but, if the force is applied on a large enough surface, the wall will move very little because of the large mass behind. A tight membrane applied to the wall can maintain its integrity after an accident. One way of ensuring this has been described previously [2].

Similar principles would have to be employed if above-ground containment vessels are to be made safe for large and rapid pressure vessel ruptures. Missiles have to be intercepted before they travel too far and blow-off routes to a large room have to be arranged for the steam. The simplest way of enclosing the reactor vessel or steam generator vessel in a second vessel as strong as the primary vessel does not seem to be a possible solution. The steam shock waves are likely to destroy the outer vessel, at least if the primary rupture is large and rapid.

Accidents such as airplane crashes, and probably sabotage and earthquakes as well, are best met by the underground solution. This is also true for war damage, but here the problem can also be solved by shutting down the plant when war threatens.

The underground location makes it possible to arrange spray and flooding systems without pumps and, in addition, large chambers are not affected by wind conditions for the decay of volatile fission products. The turbine room can be utilized in accident situations for such decay processes and discarded transport tunnels from the construction period for normal operation releases.

2. NUCLEAR SAFETY STANDARDS

2.1. Swedish safety standards today

As in most other countries, we in Sweden have modelled our safety requirements for nuclear plants on the known United States requirements. And, like the others, our authorities have added some requirements of their own.

We employed for a year the US single-failure criterion and found it an excellent guide for nuclear plant design. We find it particularly useful for judging the secondary plant systems, which may otherwise easily be overlooked. The criterion is also very useful for judging measures against secondary unrelated accidents.

Another rule which has proved very useful is one of the specific requirements imposed on us by the Swedish authorities. It is the most important requirement in a very modest package of Swedish specialities and is called the 30-minute rule. It stipulates that all actions taken within 30 minutes after an accident to avoid a major catastrophe have to be made automatic. The 30 minutes is thus the time allotted to the operator to regain his senses when he finds his plant involved in serious trouble.

Some safety questions on which ASEA-ATOM has found insufficient guidance in the published literature and on which we had to do basic work on our own standards are listed below. They are mostly in the field of external disturbances, where less work seems to have been done on standards than on the internal nuclear plant systems.

- (1) Reactor containment vessel integrity, as partly described earlier in this paper.
- (2) Internal transport accidents, which have only lately been given any place in safety requirements in other countries. ASEA-ATOM has always felt that dropping heavy objects is as dangerous as, and is also very much more probable than, sudden pipe ruptures. We have tried to design our plants to be safe for all kinds of credible transport accidents.
- (3) A close relative of the transport accident is the building collapse accident. Whereas collapse of building elements is indeed very uncommon, building codes do not generally require the kind of safety which is necessary to cope with a very big nuclear accident.
- (4) Only one safety standard has, to our knowledge, so far shown any awareness of the fact that the primary goal of fire prevention in a nuclear plant is to safeguard the safe shutdown and cooling of the reactor. We have worked hard on this problem at ASEA-ATOM, and I believe we have found a way to write our own code.
- (5) For lightning damage prevention, the case is almost identical. ASEA-ATOM has had its own standard for four years. It focuses on the safety of a redundant electrical system, where physical separation is no protection against lightning damage.

2.2. The development of nuclear safety standards

There are several reasons why the development of nuclear standards should be made into a multi-national effort. There are reasons, however, why an international effort such as that represented by the IEC and ISO is not very tempting. In our opinion, the best way would be to arrange some kind of co-operation within the existing US framework.

The major reason for us all to co-operate is that we may damage the whole of our profession if we do not. The public and the press throughout the world are extremely volatile in this area. They have often been eager to accept statements to the effect that the nuclear safety requirements in their country are stricter than those of other countries. Sweden is no exception, and neither press nor public will accept a situation in which it is stated that their safety standards are inferior. This can always be claimed for any standard, as long as there are substantial differences in the philosophical approach to safety.

We should therefore keep nuclear safety standards outside the competitive area, we should make them universally accessible as rapidly as possible, and we should try to unify them as much as possible.

The United States of America have done well in these respects. Other countries where standards have been developed have published very little, particularly on the application of the very general terms of the standards. There is thus, for instance, little in the published literature to substantiate claims that standards exist with stricter requirements than those of the United States.

Even the United States could do better, however. Like all standards, the nuclear safety standards have been slow to emerge from the committees, even where these have been established as a tool for the authorities. This can easily cause difficulties in such a problem-ridden area as that of nuclear reactors. Our experiences in Sweden in the mid-sixties may provide an example of this.

The design work on the Marviken and Oskarshamn I plants was in many ways quite parallel, with an actual separation in time of about a year. For neither type of reactor were any safety standards available in 1965. We wrote to all the US safety standards committees and requested information. Those which bothered to answer, and they were not many, mostly said that it would be some time before the "general public" would be apprised of their results. We had of course no access to the USAEC and USACRS reviewing boards, which are limited by law to judging domestic nuclear plants.

We had therefore to work on precedents. This was possible for ASEA with regard to the Oskarshamn I plant, thanks to the published USAEC decisions on US boiling-light-water reactors. The basic criteria could be deduced from these decisions and applied to our designs. When the General Design Criteria were published in 1967, Oskarshamn I conformed to them all. We only made a small change in our core spray design in order to conform to the wording of the rules, a change which actually made the system a little less efficient.

The Marviken designers of AB Atomenergi had no such decision to go on, as there were no heavy-water boilers in the United States. The fate of Marviken was quite obvious in 1967 to anyone judging the design by the General Design Criteria. If these general criteria had been accessible for the designers in 1964, the Marviken plant might have been operating today.

All this may seem to constitute a strong case for international safety standards, but it does not. In spite of what is said above, we still believe that the job can best be done for quite some time to come by the American professional societies and the USAEC. The liberal sharing of knowledge by the United States in the nuclear field, the excellent work done on the development of standards and the general abstention of US industry from utilizing in any inappropriate way their early knowledge of nuclear safety standards is an example to one and all.

A major reason why international standards work is not a tempting alternative is the notorious slow speed and occasional poor performance of such enterprises. When there is an existing open program in the leading country this should be encouraged and not played down.

Another reason for keeping the US effort as the focusing point for the world is the proximity of the people in the standards committees to the reviewing boards of the leading nation. Safety standards are not of much value if they are not supplied with rules for the application of the standards issued by a board responsible to lawmakers, a free press and the public at large.

The best possibility for fruitful international co-operation would, in our opinion, be provided by the standards committees of the US professional societies if these could accept international participation in their work.

3. NORMAL OPERATIONAL RELEASES OF RADIOACTIVE MATERIAL

The release rates of radioactive material from nuclear plants are here kept apart from the safety questions. This is, of course, only a question of definitions and does not constitute any Swedish policy.

Public pressure has caused the authorities of several countries to lower radically the permissible exposures of the nearest man below the maximum values recommended by the ICRP. Increasing and decreasing stack height and a tendency to give total release rates instead of effective rates has contributed to an even more drastic lowering of permissible release rates.

The real reason for lowering the limits of exposure of the nearest man has usually not been motivated by his exposure, however, but by the genetic implications of exposing large groups of the public to ionizing radiation.

In Sweden, development has been along different lines. It has been focused directly on the large group, and the current recommendation is that the average Swedish population exposure should not, in the long run, be more than 10 mrem/yr from all Swedish nuclear plants. This has been calculated as giving, in the year 2000, a release rate from a plant contributing a total population exposure of 1 man-rem/MW-yr of installed electrical power.

The ^{85}Kr from the fuel reprocessing plants will utilize some 5% of this dose rate. The rest will be almost entirely caused by ^{133}Xe from the power plants. As the half-life of ^{133}Xe is 5 days, however, a major part of the world will share the dose. This makes a national standard rather unreasonable, and here we have a very suitable subject for international co-operation, which is much better suited for such a venture than the safety questions dealing with accidents. This is the case because there is no necessity to judge the hardware involved and because the trans-frontier interactions are much stronger for normal operation releases than for accident releases, at least as far as total body doses are concerned.

The national pride which the public in many countries has been taught by its press to take in its "superior safety standards" could then perhaps also be circumvented. The safety most often thought of in this respect is the normal operational release-rate limitations, and the reasons for the claims are that the authorities have safeguarded themselves against public criticism by some such measure as stepping a factor of three below what "everybody else permits".

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PROBLEMES DE CORROSION POSES PAR L'EMPLOI DES ACIERS DANS LA CONSTRUCTION DES GENERATEURS DE VAPEUR POUR REACTEURS NUCLEAIRES

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Abstract-Résumé-Аннотация-Resumen

CORROSION PROBLEMS RAISED BY THE USE OF STEELS IN THE CONSTRUCTION OF STEAM GENERATORS FOR NUCLEAR REACTORS.

The steels used in the construction of steam generators include ferritic and austenitic types. In water at 350°C, ferritic steels containing 2.25% chromium are adequately corrosion resistant and are not sensitive to corrosion under stress. Austenitic "chromium-nickel-iron" alloys and steels oxidize very little, but under mechanical stress the following phenomena can appear: transgranular cracking of 18/10 type steels in the presence of chlorides; intergranular cracking of nickel-rich alloys, particularly structurally hardened types, in pure water; absence of cracking in alloys of intermediate nickel content, between 30 and 40%. In addition, a study of the corrosion of austenitic alloys in steam at 600°C has shown the favourable effect on oxidation of additive elements: nickel, chromium, molybdenum, silicon. Here again, a sensitivity to intergranular cracking under stress is observed in nickel-rich alloys. Mention is made of industrial experience acquired with these materials. In a flow of liquid sodium the long-term behaviour of various ferritic steels containing between 2 and 9% chromium, and of non-alloyed steels or austenitic steels, are compared between 475 and 650°C. At 475°C, stabilized or non-stabilized ferritic steels suffer negligible weight losses, show no appreciable decarbonization and keep their original mechanical properties. Ordinary steels are strongly decarbonized under the same conditions. At 650°C, on the other hand, non-stabilized 2% chromium steel shows considerable decarbonization. This phenomenon does not affect 9% chromium steel, which presents a degree of corrosion lower than 0.5 mg/cm² after 6000 hours' treatment, similar to that of austenitic steels. Finally, all the steels suffer a limited preferential attack at the grain boundaries.

PROBLEMES DE CORROSION POSES PAR L'EMPLOI DES ACIERS DANS LA CONSTRUCTION DES GENERATEURS DE VAPEUR POUR REACTEURS NUCLEAIRES.

Les aciers utilisés dans la construction des générateurs de vapeur comprennent des nuances ferritiques et austénitiques. Dans l'eau à 350°C, les aciers ferritiques à 2,25% de chrome présentent une résistance à la corrosion suffisante et ne sont pas sensibles à la corrosion sous contrainte. L'oxydation des aciers et alliages austénitiques chrome-nickel-fer est très faible, mais lorsqu'ils sont soumis à des contraintes mécaniques, ces matériaux peuvent présenter les phénomènes suivants: fissuration transgranulaire des aciers du type 18/10 en présence de chlorures; fissuration intergranulaire des alliages riches en nickel, et, particulièrement, des nuances à durcissement structural, dans l'eau pure; absence de fissuration des alliages à teneurs en nickel intermédiaires, comprises entre 30 et 40%. D'autre part, l'étude de la corrosion dans la vapeur d'eau à 600°C des alliages austénitiques a permis de préciser l'influence bénéfique sur l'oxydation des éléments d'addition: nickel, chrome, molybdène, silicium. Dans ce domaine, on constate également une sensibilité à la fissuration intergranulaire sous contrainte des alliages à haute teneur en nickel. Les auteurs font état d'expériences industrielles acquises sur ces différents matériaux. Dans le sodium liquide en circulation, ils comparent, entre 475 et 650°C, les comportements à long terme de divers aciers ferritiques à teneurs en chrome comprises entre 2 et 9% et d'aciers non alliés ou d'aciers austénitiques. A 475°C, les aciers ferritiques stabilisés ou non subissent des pertes de poids pratiquement

négligeables, ne présentent pas de décarburation notable et conservent leurs caractéristiques mécaniques initiales. Dans les mêmes conditions, les aciers ordinaires sont sévèrement décarbures. Par contre, à 650°C, l'acier à 2% de chrome, non stabilisé, subit une décarburation considérable. Ce phénomène n'affecte pas l'acier à 9% de chrome qui présente un taux de corrosion inférieur à 0,5 mg/cm² après 6000 heures de traitement, analogue à celui des aciers austénitiques. Enfin, tous les aciers subissent une attaque préférentielle, quoique limitée, des joints de grains.

КОРРОЗИОННЫЕ ПРОБЛЕМЫ, ВОЗНИКАЮЩИЕ ПРИ ИСПОЛЬЗОВАНИИ СТАЛЕЙ В КОНСТРУКЦИЯХ ПАРОГЕНЕРАТОРОВ ДЛЯ ЯДЕРНЫХ РЕАКТОРОВ.

Материалы, используемые в конструкциях парогенераторов, включают ферритные и аустенитные стали. В воде при 350°C ферритные стали, содержащие 2,25% хрома, достаточно коррозионностойки и не чувствительны к коррозии под напряжением. Аустенитные сплавы "хром-никель-железо" и аустенитные стали окисляются очень слабо, однако в условиях механических напряжений для них характерны следующие явления: транскристаллитное растрескивание сталей типа 18/10 в присутствии хлоридов; межкристаллитное растрескивание в чистой воде богатых никелем сплавов, особенно структурно упрочняющихся сплавов; отсутствие растрескивания в сплавах, содержащих 30-40% никеля. Кроме того, изучение коррозии аустенитных сплавов в паре при 600°C показало, что легирующие элементы — никель, хром, молибден и кремний — оказывают благотворное влияние на стойкость к окислению. В этом случае богатые никелем сплавы также оказались чувствительными к межкристаллитному растрескиванию под напряжением. Приводятся данные, касающиеся промышленного опыта, накопленного по указанным различным материалам. Сравнивается поведение различных ферритных сталей, содержащих 2 и 9% хрома и нелегированных сталей или аустенитных сталей в процессе длительных испытаний в потоке жидкого натрия при температурах 475-650°C. При 475°C стабилизированные и нестабилизированные ферритные стали обнаруживают небольшие весовые потери, несущественное обезуглероживание и сохраняют исходные механические свойства. Обычные стали сильно обезуглероживаются в этих условиях. С другой стороны, при 650°C нестабилизированная сталь, содержащая 2% хрома, заметно обезуглероживается. Сталь, содержащая 9% хрома, не обезуглероживается, причем скорость коррозии этой стали составляет менее 0,5 мт/см² после испытания в течение 6000 ч (поведение аустенитных сталей аналогично). Все стали обнаруживали небольшую преимущественную коррозию по границам зерен.

PROBLEMAS DE CORROSION PLANTEADOS POR EL EMPLEO DEL ACERO COMO MATERIAL DE CONSTRUCCION EN LOS GENERADORES DE VAPOR DE LOS REACTORES NUCLEARES.

Los aceros utilizados en la construcción de los generadores de vapor, presentan matices diferentes entre los tipos ferríticos y austeníticos. En el agua a 350°C, los aceros ferríticos con 2,25% de cromo tienen una gran resistencia a la corrosión y, además, ésta no varía si el acero está sometido a tensiones. La oxidación de los aceros y aleaciones austeníticas cromo-níquel-hierro es muy débil, pero cuando están sometidos a tensiones mecánicas, pueden presentar los fenómenos siguientes: grietas intragranulares en los aceros del tipo 18/10, en presencia de cloruros; grietas intergranulares en las aleaciones ricas en níquel, a la vez que, particularmente, en agua pura presentan una especie de endurecimiento estructural; ausencia de grietas en las aleaciones con un contenido de níquel comprendido entre el 30 y el 40%. Por otra parte, el estudio de la corrosión de las aleaciones austeníticas, en un medio de vapor de agua a 600°C, ha permitido precisar la influencia beneficiosa que sobre la oxidación ejerce la adición de los elementos: níquel, cromo, molibdeno y silicio. En iguales condiciones, se ha comprobado una sensibilidad a la formación de grietas intergranulares en las aleaciones con elevado contenido en níquel, cuando están bajo tensiones. También se hace referencia a la experiencia industrial que existe sobre estos materiales. A temperaturas comprendidas entre 475 y 650°C, y en un medio de sodio líquido en circulación se compara el comportamiento a largo plazo de los diversos aceros ferríticos con un contenido en cromo comprendido entre 2 y 9% y de los aceros no aleados o de los aceros austeníticos. Los aceros ferríticos estabilizados o no, sufren una pérdida de peso prácticamente despreciable a la temperatura de 475°C. No presentan descarburación notable y conservan sus características mecánicas iniciales. No obstante, en las mismas condiciones, los aceros ordinarios presentan una gran descarburación. Por el contrario, a 650°C, el acero con 2% de cromo, no estabilizado, sufre una descarburación considerable. Este fenómeno no afecta al acero con 9% de cromo, el cual presenta un índice de corrosión inferior a 0,5 mg/cm² después de 6000 horas de tratamiento. Este valor es parecido al de los aceros austeníticos. Finalmente, se puede decir que todos los aceros sufren un ataque, si bien limitado, preferentemente en los bordes del grano.

Les matériaux utilisés dans la construction des générateurs de vapeur des réacteurs nucléaires comprennent des alliages austénitiques "chrome, nickel, fer" et des aciers ferritiques. Nous nous proposons d'étudier dans ce mémoire, à partir d'essais récents et d'expériences industrielles, un certain nombre de problèmes liés aux réacteurs à eau et aux réacteurs à neutrons rapides refroidis au sodium.

1. CORROSION SOUS CONTRAINTE DANS L'EAU ET LA VAPEUR A HAUTE TEMPERATURE

Les aciers inoxydables austénitiques, tels que ceux du type Cr 18%, Ni 10% sont, comme il est bien connu, sensibles à la fissuration sous contrainte, particulièrement dans les milieux chlorurés. Les travaux de COPSON [1] ayant montré que les alliages "chrome, nickel, fer" à teneur élevée en nickel, du type de l'Inconel 600, présentaient une excellente résistance à la corrosion sous tension dans la solution classique de chlorure de magnésium bouillant à 154°C, l'Inconel 600 a été largement utilisé pour la construction de générateurs de vapeur sur les P.W.R. Toutefois, au cours de diverses études [2 à 4], on a mis en évidence que les alliages austénitiques "chrome, nickel, fer", à teneur élevée en nickel (Ni = 75-77%), par exemple l'Inconel 600, sont sensibles à un phénomène particulier de corrosion sous contrainte. La fissuration se révèle essentiellement intergranulaire et apparaît dans l'eau déminéralisée (350°C) ou la vapeur surchauffée (600°C), ceci en l'absence de traces d'impuretés nocives (plomb), de zones de recoin [5] et indépendamment de la teneur en oxygène et en chlorures, du moins pour des alliages à l'état non sensibilisé. Des essais, récemment effectués aux U.S.A. [6], dans l'eau pure à 350°C, confirment nos résultats. A noter que la période d'incubation précédant l'apparition des fissures est longue (généralement de l'ordre de plusieurs mois).

Les figures 1 et 2 mettent en évidence l'aspect micrographique du phénomène, d'une part sur un alliage Cr 17%, Ni 77%, à teneur en carbone ultra basse (C = 0,002%), d'autre part sur un alliage commercial Inconel 600 (C = 0,040%). Ces deux alliages, à l'état décapé après mise en solution par trempe, sous une contrainte exercée par flexion (déformation permanente de 0,5% mesurée à froid), ont été soumis à l'action de l'eau déminéralisée et désoxygénée ($O_2 \leq 0,003$ mg/l). Dans les mêmes conditions, mais avec une addition de chlorure (1 g/l de Cl^- sous forme de NaCl), on a en outre expérimenté une série d'alliages austénitiques "chrome, nickel, fer" à 17% de chrome et à teneur en nickel variant de 10 à 77%. Après six mois de corrosion, on a constaté l'apparition de fissures sur les alliages à 10% et 77% de nickel, les fissures étant transgranulaires sur l'alliage à 10% de nickel et intergranulaires sur l'alliage à 77% de nickel. Des amorces de fissuration transgranulaire ont également été relevées sur l'alliage à 15% de nickel. Les alliages à 25, 35, 45 et 65% de nickel ne présentaient, par contre, aucun phénomène particulier. Ces essais montrent donc que la teneur en nickel de l'alliage est le facteur prédominant qui détermine l'apparition et le type de la fissuration.

Sur le plan pratique, l'ensemble des résultats exposés met en évidence l'intérêt des alliages austénitiques "chrome, nickel, fer" à teneurs moyennes en nickel (Ni = 35 à 45%).



FIG. 1. Alliage Cr 17%, Ni 77% fissuré après six mois d'essai dans l'eau à 350° C ($O_2 \leq 0,003$ mg/l).

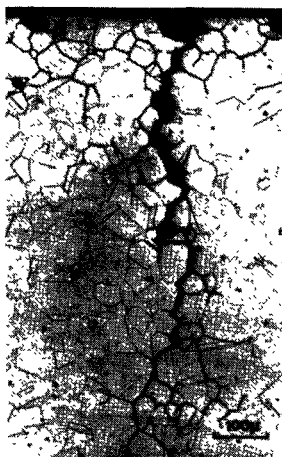
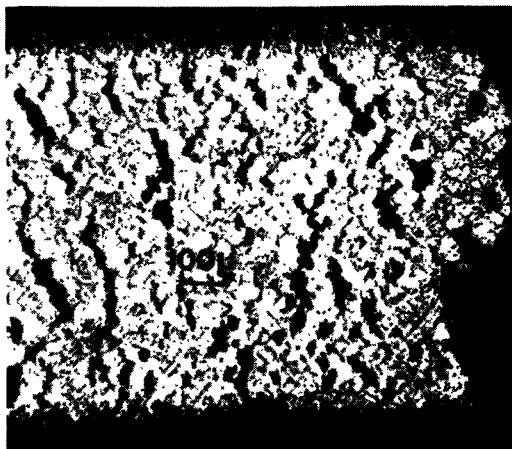


FIG. 2. Alliage Inconel 600 fissuré après neuf mois d'essai dans l'eau à 350° C ($O_2 \leq 0,003$ mg/l).

On a d'autre part récemment étudié le comportement sous contrainte de l'alliage Inconel X 750 dans l'eau déminéralisée et désoxygénée ($O_2 \leq 0,003$ mg/l), en circulation, à 300 et 350°C. Les échantillons, prélevés dans de la barre, de la plaque ou de la tôle, préparés par usinage, abrasion ou décapage chimique, étaient contraints par traction ou flexion entre $2/3 E_{0,2}$ ($E_{0,2}$ mesuré à la température de l'essai) et 0,5% de déformation permanente à la température ambiante. On a constaté que l'alliage, à l'état durci par précipitation, se révèle très sensible à la fissuration intergranulaire sous contrainte : la période d'incubation est beaucoup plus courte que celle relative à l'alliage non durcissable. Bien que le phénomène de fissuration intergranulaire ait été relevé pour les



a)



b)

FIG. 3. Alliage Cr 17%, Ni 77%: a) fil rompu dans la vapeur d'eau à 600° C (100 kg/cm², O₂ ≤ 0,003 mg/kg) (contrainte appliquée par traction: 6,5 kg/mm²); b) fil témoin traité sous argon à 600° C (contrainte appliquée par traction: 6,5 kg/mm²).

différents états de surface considérés, les échantillons ayant subi un décapage chimique, même si celui-ci ne conduit pas à des indentations intergranulaires, sont plus sensibles à la fissuration que les échantillons préparés par abrasion. Le traitement thermique est un facteur très important : dans l'état de mise en solution, l'alliage se comporte comme la nuance non durcissable. La sensibilité à la fissuration augmente avec le niveau de précipitations et donc de propriétés mécaniques.

A l'aide d'un dispositif expérimental, spécialement mis au point, caractérisé par une charge extérieure et un passage étanche à travers

l'autoclave, on a étudié, comparativement, dans la vapeur à 600° C (100 bars, $O_2 \leq 0,003$ mg/kg), la résistance à la corrosion sous contrainte des alliages "chrome, nickel, fer" : Cr 17 %, Ni 77 %, (C = 0,016 %) et Cr 17 %, Ni 10 %, (C = 0,027 %) . Les échantillons se présentaient sous la forme de fil ($\phi = 2$ mm), à l'état décapé après mise en solution par trempe ; la charge appliquée était de 6,5 kg/mm². Tous les fils de l'alliage Cr 17 %, Ni 77 % se sont rompus entre cinq et six mois. Des échantillons témoins des mêmes fils, soumis sous atmosphère d'argon aux mêmes conditions de contrainte, de température et de durée, n'ont par contre donné lieu à aucun phénomène particulier. Ce résultat prouve donc que les ruptures observées dans la vapeur à 600° C correspondent bien à un phénomène de corrosion sous contrainte. Sur les fils rompus, les fissures sont intergranulaires (FIG. 3). Les fils en acier inoxydable (Cr 17 %, Ni 10 %) n'ont donné lieu à aucune rupture pour des essais poursuivis jusqu'à 14 mois.

Les résultats rapportés précédemment et des cas récents de fissuration intergranulaire sous contrainte de tubes d'échangeurs en Inconel 600, montés sur des réacteurs actuellement en service, mettent en doute la sécurité présentée par les alliages de ce type.

2. OXYDATION DANS L'EAU ET LA VAPEUR A HAUTE TEMPERATURE

Dans l'eau désoxygénée ($O_2 \leq 0,003$ mg/l), à 350° C, les aciers ferritiques classiques à 2,25 % de chrome donnent lieu à une oxydation uniforme acceptable, le métal se recouvrant d'une couche de magnétite relativement protectrice. Il convient toutefois de signaler que leur comportement est moins satisfaisant dans l'eau à basse température et, en cas d'arrêt des installations, il est nécessaire de prendre des précautions particulières pour éviter la formation d'hydroxydes non adhérents. Les aciers ferritiques présentent naturellement, par rapport aux aciers austénitiques, l'avantage de ne pas être sensibles à la corrosion sous contrainte ; ils peuvent cependant conduire à l'introduction de produits solides en suspension dans les circuits primaires des réacteurs, d'où une activation susceptible de se révéler importante.

Dans l'eau à 350° C, l'oxydation des aciers inoxydables austénitiques est relativement faible. Toutefois, dans la vapeur au-delà de 500° C, celle-ci s'accroît notablement et il paraît intéressant de rechercher des alliages présentant un comportement amélioré [7]. On a donc expérimenté les gammes suivantes d'alliages "chrome, nickel, fer", à l'état décapé après mise en solution par trempe :

1. Cr 17 %, Ni 10 à 77 %
2. Cr 17 %, Ni 15 %, Si 0,02 à 5 %
3. Cr 17 %, Ni 15 %, Mo 1 à 5 %
4. Cr 17 %, Ni 15 %, Si 0,5 à 2,5 % et Mo 0,5 à 2,5 %
5. Cr 10 à 25 %, Ni 25 %.

Dans tous les cas, les teneurs en carbone sont inférieures à 0,03 %.

Après quatre mois d'essai dans la vapeur à 600° C (100 bars, $O_2 \leq 0,003$ mg/kg), ces différents alliages ont conduit aux résultats illustrés par les figures 4 à 8. La corrosion est exprimée en perte de métal après élimination sélective des produits d'oxydation.

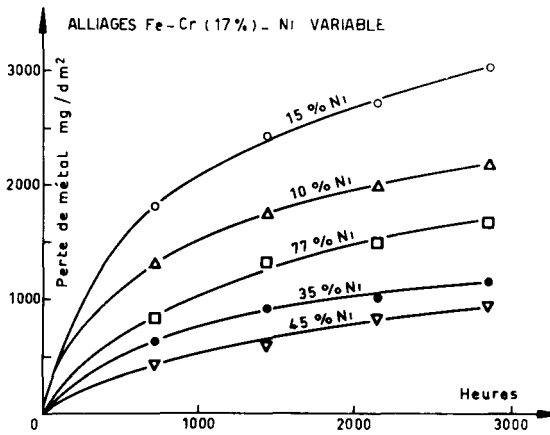


FIG. 4. Essais dans la vapeur d'eau surchauffée: Ni variable (température 600° C, pression 100 bars, $O_2 \leq 0,003$ mg/kg).

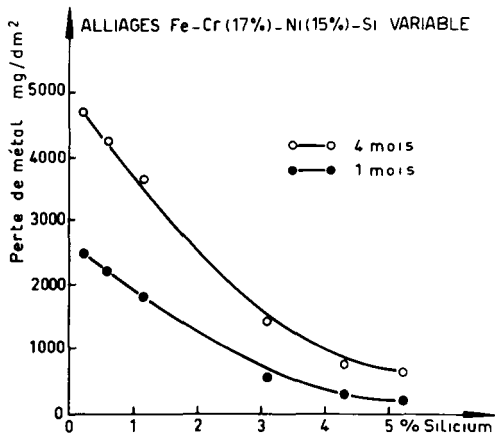


FIG. 5. Essais dans la vapeur d'eau surchauffée: Si variable (température 600° C, pression 100 bars, $O_2 \leq 0,003$ mg/kg).

On note que chacun des éléments : nickel, silicium, molybdène, chrome, a une influence bénéfique sur l'oxydation. En ce qui concerne les additions de nickel, la corrosion la plus faible est obtenue pour une teneur de l'ordre de 35 à 45 % (FIG. 4).

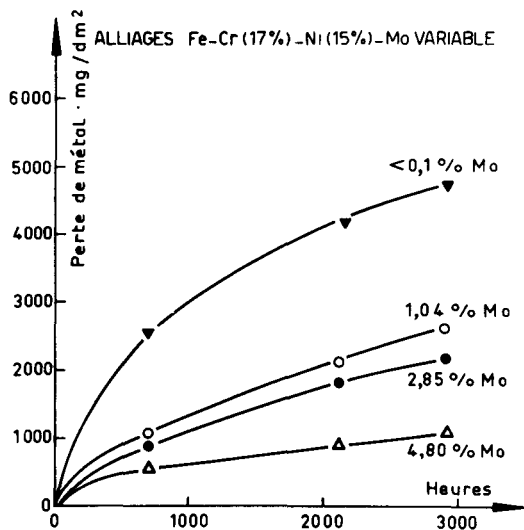


FIG. 6. Essais dans la vapeur d'eau surchauffée: Mo variable (température 600° C, pression 100 bars, $O_2 \leq 0,003$ mg/kg).

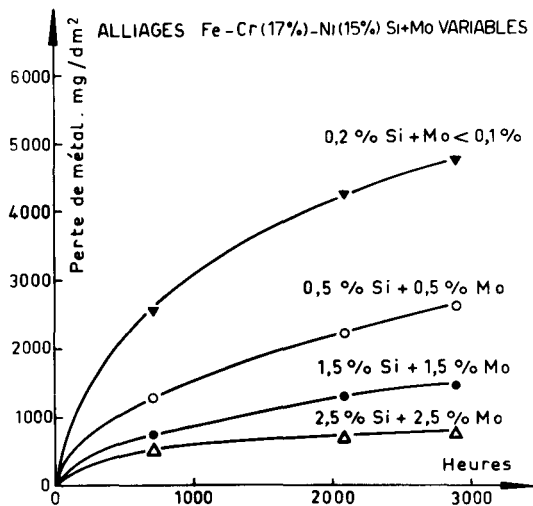


FIG. 7. Essais dans la vapeur d'eau surchauffée: Si + Mo variables (température 600° C, pression 100 bars, $O_2 \leq 0,003$ mg/kg).

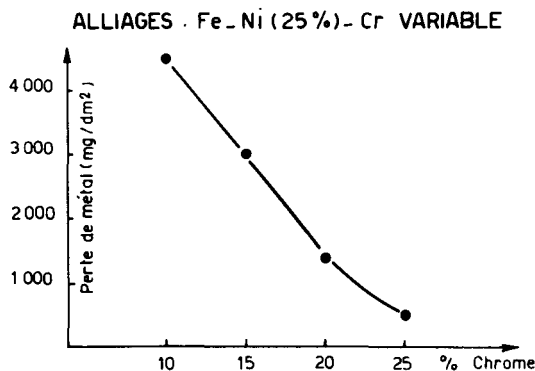


FIG. 8. Essais dans la vapeur d'eau surchauffée: Cr variable (température 600° C, pression 100 bars, $O_2 \leq 0,003$ mg/kg).

3. EXPERIENCE INDUSTRIELLE DE L'UTILISATION DES ALLIAGES "CHROME, NICKEL, FER" DANS L'EAU ET LA VAPEUR A HAUTE TEMPERATURE

Les alliages "chrome, nickel, fer" et plus particulièrement les aciers inoxydables austénitiques du type Cr 18 %, Ni 10 % sont utilisés depuis de longues années dans la construction de chaudières et d'échangeurs de chaleur ; en règle générale, ces appareillages donnent satisfaction aux usagers. Toutefois, en cas de contamination du milieu, des incidents peuvent se produire.

L'acier Cr 18 %, Ni 10 %, stabilisé au titane (AISI TP 321) est exclusivement utilisé en France dans les chaudières de centrales électriques pour la construction des parties chaudes des surchauffeurs de vapeur. Des chaudières, avec des caractéristiques de vapeur de 565° C et 140 bars, ont atteint 50 000 heures de service sans qu'il ait été nécessaire de procéder à une intervention.

Toujours dans le domaine de la vapeur à haute température, l'alliage Incoloy 800 (Cr 20 %, Ni 32 %) a été mis en oeuvre pour construire un surchauffeur élevant la température de la vapeur (32 bars) de 275° C à 650° C. L'appareil a 40 000 heures de service sans incident.

Des dizaines d'échangeurs de chaleur en acier inoxydable austénitique fonctionnent ainsi depuis plusieurs années sans soucis pour les utilisateurs. Néanmoins, quelques difficultés rencontrées en service mettent en évidence la concomitance des paramètres qui conduisent à la détérioration de ces alliages et la difficulté d'en faire une approche quantitative par la seule expérience industrielle.

Bien que la possibilité de corrosion par piqure ne soit pas à écarter, la corrosion fissurante a été le dommage le plus fréquemment rencontré, et qui a toujours mis le matériel hors service.

La nature du milieu agressif et sa concentration en produits nocifs, la température et les contraintes internes ou externes auxquelles est soumis le matériau, concourent à l'apparition parfois très rapide de la fissuration des aciers inoxydables austénitiques : il peut suffire qu'un seul de ces facteurs soit absent pour que le matériau reste presque indéfiniment en service sans dommage.

Nous citerons quelques exemples. On connaît la réputation d'agressivité des chlorures et de la soude vis-à-vis des aciers inoxydables austénitiques ; or, il a été construit en 1965 des chaudières à tubes de fumée en acier 18/10, à très bas carbone (304 L) qui fonctionnent depuis plus de cinq ans en continu avec une teneur en chlorure de sodium de l'eau de chaudière supérieure à 1 g/l; le pH est de 11,8 (additions de soude caustique et de phosphate). La température de l'émulsion atteint 132° C.

Dans la même installation, un réchauffeur d'eau en forme de serpent construit en acier du type Cr 18 %, Ni 12 %, Mo 2 à 3 %, à très bas carbone (316 L), fonctionne sans ennui à une température allant de 104° C (entrée) à 204° C (sortie) ; l'eau de pH 9 à 10 contient plus de 100 mg/l de NaCl.

Il n'en a pas été de même pour un générateur de vapeur utilisé sur un complexe chimique à 116 bars, avec une température d'émulsion de 322° C ; l'eau, de haute pureté, est traitée à pH 9 par de l'ammoniaque. Les plaques tubulaires en acier 18/10 stabilisé au titane (321) et les tubes de même nuance sont assemblés par la technique classique du dudgeonnage et de la soudure d'étanchéité ; la formation du côté eau d'une petite cavité entre le tube et la plaque est inévitable. La chaudière a fonctionné parfaitement jusqu'à ce qu'une fausse manoeuvre au poste de préparation d'eau ait amené dans la chaudière une quantité assez importante de chlorure et de soude caustique. Cela a provoqué sur le champ une fissuration transgranulaire des tubes dans l'extrémité de leur partie dudgeonnée, au contact de l'eau.

Cet incident met nettement en évidence l'influence simultanée de la déformation permanente du métal par le dudgeonnage, de l'effet de crevasse apporté par la cavité, et de la température : déformation du métal et crevasse concourent en particulier à provoquer la fissuration, les parties libres du tube n'étant en effet pas fissurées. Il est d'ailleurs fort possible que rien d'anormal ne se serait produit si la température avait été plus basse, puisque la chaudière dont il a été fait état précédemment fonctionne en continu, depuis cinq ans, avec de l'eau à 132° C et dans des conditions permanentes de salinité vraisemblablement aussi sévères que celles qui, dans le cas présent (322° C), ont provoqué des fissurations en quelques heures. Détail encore plus convaincant, les techniques d'assemblage des tubes et des plaques sont très voisines dans les deux chaudières.

4. CORROSION DES ACIERS DANS LE SODIUM

Les aciers inoxydables austénitiques classiques sont maintenant couramment utilisés comme matériaux constitutifs des circuits à sodium liquide jusqu'à des températures de l'ordre de 650° C. Cependant l'intérêt économique qui s'attacherait à leur remplacement, même partiel, par des aciers ferritiques, est évident.

L'un des problèmes majeurs posés par l'utilisation de ces aciers ferritiques, en présence de sodium liquide, réside dans leur tendance à la décarburation. Ce phénomène, préjudiciable à leurs caractéristiques mécaniques, pourrait devenir particulièrement gênant si des aciers austénitiques se trouvaient utilisés conjointement : ces derniers étant alors susceptibles de subir une carburation conduisant à une altération de leurs propriétés mécaniques.

TABLEAU I. CONDITIONS DE FONCTIONNEMENT DU CIRCUIT A SODIUM

Température des échantillons (°C)	475 ou 650
Température de la zone froide (°C)	300
Débit du sodium (l/s)	2
Vitesse sur les échantillons (m/s)	3-2-1, 3-0,4
Débit du piège froid (l/s) sur circuit dérivé	0,15
Température du piège froid (°C)	105 - 110
Teneur en oxygène (ppm)	5 - 10
Teneur en carbone libre (ppm)	10 - 15
Teneur en carbonate (ppm)	5 - 10

TABLEAU II. COMPOSITION DES ACIERS ETUDIES DANS LE SODIUM, EXPRIMEE EN POUR-CENT

Désignation AFNOR	C	Si	Mn	Cr	Ni	Mo	V	Nb
A 42	0,15	0,14	0,63	-	-	-	-	-
20 D2	0,19	0,34	1,30	0,16	0,6	0,5	-	-
10 CD V9	0,10	0,31	0,50	2,37	-	0,9	0,11	-
7 CD Nb 9	0,07	0,25	0,69	2,23	-	1,0	-	1,02
10 CD Nb V10	0,11	0,30	0,50	2,50	0,2	0,9	0,50	0,50
Z10 CD Nb V9	0,07	0,20	1,15	9,7	-	1,4	0,87	0,13
Z2 CN 18-10	0,024	0,41	1,73	18,2	9,8	-	-	-
Z2 CND 17-12	0,021	0,45	1,78	17,6	13,0	1,1	-	-

La décarburation des aciers ferritiques à 2,25 % de chrome apparaît au-dessus de 450-500° C [8 à 10]. Elle semble toutefois pouvoir être limitée et même annulée par des additions de niobium, titane ou vanadium. Par ailleurs, les nuances plus chargées en chrome -5 à 12 %- ne sont pratiquement pas affectées jusqu'à 600° C [9] [10].

En ce qui concerne les aciers inoxydables, leur bonne tenue jusqu'à 650° C dans le sodium contenant 10 à 20 ppm d'oxygène est admise d'une façon générale [11] [12]. Le comportement de ces aciers dans le sodium liquide est néanmoins étroitement lié au degré de pureté de ce dernier [13 à 15].

Nous résumerons ici les résultats des essais que nous avons effectués dans des conditions de purification et de vitesse du sodium au niveau des échantillons voisines de celles rencontrées dans les réacteurs (2 à 3 m/s) (Tableau I). La composition des différents aciers étudiés figure dans le Tableau II. Le contrôle analytique du sodium est réalisé au moyen d'un indicateur de bouchage dont la température se maintient entre 100 et 105° C, exception faite des périodes de chargement. Ces indications sont complétées par des analyses d'oxygène et de carbone effectuées régulièrement sur des prélèvements de sodium [16] [17].

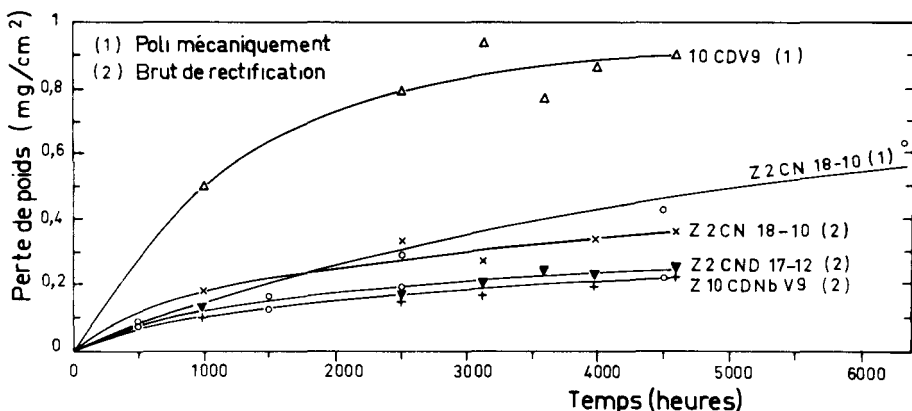


FIG. 9. Cinétiques de corrosion des aciers austénitiques et ferritiques dans le sodium à 650°C . Vitesse du sodium: 2 m/s.

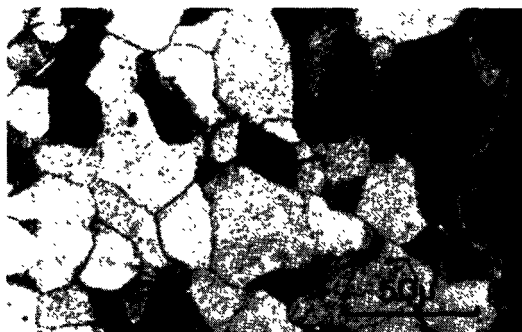


FIG. 10. Examen sur coupe de l'acier 10 CDV 9 après exposition de 6000 heures dans le sodium à 475°C .

4. 1. Cinétiques de corrosion

A 475°C , les aciers ferritiques subissent des pertes de poids très faibles, puisque après 10 000 heures d'exposition elles représentent des pertes d'épaisseur inférieures au micron. La vitesse de dissolution du métal devient même pratiquement nulle après 1 000 à 2 000 heures d'essai.

Si l'influence de la vitesse de circulation du sodium apparaît négligeable, par contre l'état de surface des aciers intervient nettement dans la cinétique de corrosion, du moins durant les tout premiers milliers d'heures : c'est ainsi que, comparé à un état poli mécaniquement, l'état décapé-passivé conduit à des pertes de poids deux à trois fois supérieures.

A 650°C , les pertes de poids subies par les aciers ferritiques et austénitiques sont également très faibles. Les courbes correspondantes (FIG. 9) ont une allure générale parabolique, mais on peut également considérer qu'après une période initiale de 1 000 à 2 000 heures la cinétique devient quasiment linéaire.

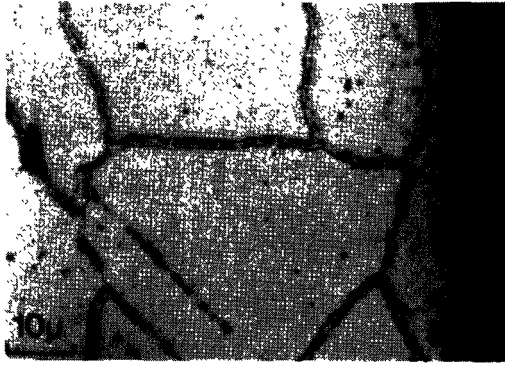


FIG. 11. Examen sur coupe de l'acier Z2 CND 17-12 après exposition de 3000 heures dans le sodium à 650 ° C.

4.2. Transfert de carbone

Bien que les aciers très peu ou non alliés soient sévèrement décarburés à 475° C, aucun des aciers ferritiques traités à cette même température ne révèle de décarburation visible due au sodium. C'est ainsi que la structure mixte "ferrite-phase riche en carbone" de l'acier à 2,25% de chrome non stabilisé n'apparaît pas modifiée après 6 000 heures d'exposition dans le sodium (FIG. 10) : les grains carburés qui se trouvaient au contact du sodium ne semblent pas attaqués, ce qui permet de penser que cette phase est encore stable à 475° C en présence de sodium. Quant aux nuances stabilisées, leur structure est caractérisée par une fine précipitation et ne se trouve pas modifiée par le traitement dans le sodium liquide.

Par contre, à 650° C, l'acier à 2,25 % de chrome non stabilisé est presque totalement décarburé dès 1 000 heures d'exposition, alors que l'acier à 9 % de chrome révèle une résistance excellente à cet égard, puisqu'on observe seulement dans cet alliage une fine précipitation de carbures, sans gradient entre la surface et le cœur.

Les aciers austénitiques ne subissent, dans les mêmes conditions, aucune carburation notable, comme le montrent les examens micrographiques et les analyses de carbone. On constate uniquement l'apparition, à 650° C, d'une zone superficielle d'une dizaine de microns d'épaisseur, appauvrie en nickel, ainsi que la présence de précipités de carbures. Ceux-ci, dans le cas de l'acier Z2 CND 17-12, sont localisés aux joints de grains qui se trouvent fortement marqués (FIG. 11). Toutefois, ce dernier phénomène ne semble pas imputable au sodium ou à ses impuretés, puisqu'il apparaît également sur des échantillons simplement traités sous vide pendant la même durée et à la même température.

4.3. Phénomènes superficiels

A 650° C, on constate une modification de l'état de surface de tous les aciers, provoquée par un phénomène de dissolution préférentielle.



FIG. 12. Aspect superficiel de l'acier Z2 CN 18-10 après exposition de 6000 heures dans le sodium à 650° C.

TABLEAU III. COMPARAISON DES CARACTERISTIQUES MECANQUES: à l'état initial (premier nombre) et après traitement à 650°C, pendant 1000 heures dans le sodium (deuxième nombre) ou sous vide (troisième nombre)

Nuance d'acier	Charge de rupture R (kg/mm ²)	Allongement A (%)	Résilience K _{UF} (kg.m/cm ²)
10 CD V9	70 - 49 - 50	18 - 26 - 26	8 - 12 - 11
Z10 CD Nb V9	87 - 68 - 69	10 - 17 - 16	0,5 - 0,5 - 0,5
Z2 CN 18-10	58 - 58 - 58	62 - 60 - 58	12 - 13 - 14
Z2 CND 17-12	57 - 56 - 57	50 - 49 - 47	12 - 12 - 11

Cela se traduit par une disparition progressive des stries d'usinage et également par une attaque sélective des joints de grains, avec effet de gravage (FIG. 12).

4.4. Evolution des propriétés mécaniques

Après traitement dans le sodium à 475° C, les caractéristiques mécaniques des aciers ferritiques à 2,25 % de chrome, stabilisés ou non, restent pratiquement inchangées, même après 10 000 heures d'exposition.

Par contre, les aciers ferritiques ayant séjourné dans le sodium à 650° C révèlent une évolution relativement sensible de leurs caractéristiques mécaniques ; mais cette dernière, particulièrement marquée pendant les premiers milliers d'heures, paraît encore résulter du traitement thermique, plutôt que de l'action propre du sodium (Tableau III).

5. CONCLUSION

Nous avons essayé d'analyser dans ce mémoire un certain nombre de problèmes de corrosion concernant les générateurs de vapeur pour réacteurs nucléaires.

Ainsi que nous l'avons vu, la forme de corrosion la plus à craindre dans l'eau ou la vapeur à haute température est la fissuration sous contrainte des alliages austénitiques. En dehors de la fissuration transgranulaire classique des aciers inoxydables austénitiques en milieu chloruré, on voit apparaître un phénomène particulier de fissuration intergranulaire sous contrainte pour les alliages à teneur élevée en nickel, tels que ceux du type de l'Inconel 600. Des alliages austénitiques à teneur moyenne en nickel (Ni : 35 à 45 %) devraient permettre de donner une solution intéressante au problème. D'autre part, on montre que divers éléments d'addition (nickel, silicium, molybdène, chrome) ont une influence importante sur la cinétique d'oxydation des alliages austénitiques dans la vapeur à 600 ° C (100 bars). Ces résultats permettent donc de mieux guider le choix des matériaux.

En ce qui concerne la corrosion par le sodium, le comportement des aciers ferritiques à 2,25 % de chrome stabilisés ou non est très satisfaisant à 475 ° C : ils subissent des pertes de poids très faibles et ne présentent pratiquement pas de décarburation, ni de variation de propriétés mécaniques. On confirme d'autre part l'excellente tenue des aciers austénitiques jusqu'à 650 ° C, tant du point de vue de leurs pertes de poids -très réduites- que de celui de leurs caractéristiques mécaniques. A cette même température, l'acier à 2,25 % de chrome subit une décarburation sévère, mais le comportement de l'acier renfermant 9 % de chrome se révèle satisfaisant. La présence de nickel n'est donc pas indispensable pour assurer une bonne résistance à la corrosion dans le sodium liquide, pourvu que la teneur en chrome soit suffisante.

Enfin, tous ces aciers présentent, à 650 ° C, un phénomène de "gravage" superficiel, leurs joints de grains se trouvant préférentiellement attaqués et révélés de façon spectaculaire. Ce phénomène mineur, qui n'affecte que l'épiderme métallique, n'est certainement pas gênant du point de vue pratique ; toutefois, ses conséquences sur les caractéristiques de frottement d'organes mobiles seraient à préciser.

REMERCIEMENTS

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К ВОПРОСУ О БЕЗОПАСНОСТИ КОРПУСОВ ДАВЛЕНИЯ АТОМНЫХ РЕАКТОРОВ С ТОЧКИ ЗРЕНИЯ ХРУПКОГО РАЗРУШЕНИЯ

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Abstract—Résumé—Аннотация—Resumen

SAFETY OF REACTOR PRESSURE VESSELS FROM THE STANDPOINT OF BRITTLE FAILURE.

To evaluate the safety of reactor pressure vessels use is made of a fracture analysis diagram, i.e. a plot of the tensile strength, yield point and brittle failure against temperature and the crack arrest temperature (CAT) against strength. The fracture analysis diagram together with knowledge of the stress field and size of the flaws in the material composing the pressure vessel can be used to determine the lowest permissible temperature for a given load, or else to determine the parameters (pressure and temperature) during the overpressure test and under start-up conditions when the reactor is in operation.

The fracture analysis diagram is based on the dependence of the tensile strength and yield point on temperature, as determined with standard specimens (taking into account the size effect in accordance with our previous work) and on the dependence of the crack arrest temperature on stress in natural-size specimens at two stress levels (two-thirds of the yield point and the yield point itself). Brittle failure is then determined on the basis of ASTM recommendations; WOL-X specimens are advantageous in view of the minimum dimensions required.

The fracture analysis diagram may be affected not only by the size effect, which we take into account, but also by the accumulated energy, aging of the material and radiation damage. In order to study the extent to which the first of these factors is influential, we carried out crack arrest tests on mechanically loaded plates and pressure vessels containing water or air under pressure. The tests showed that, provided no abnormally large cracks appear in the pressure vessel, the crack arrest temperature criterion ensures adequate resistance to brittle failure.

The effect of neutron irradiation shows up primarily as an increase in the strength parameters (yield point and tensile strength) and in a reduction in ductility associated with embrittlement, which is characterized by a higher brittle transition temperature and crack arrest temperature. Variation in these parameters depends greatly on the initial state of the material and on its microstructure, as has been demonstrated experimentally. There is a simultaneous decrease in the impact strength parameters in the ductile region.

To study the effect of radiation damage on the properties of the material under actual conditions use is made of a program of surveillance specimens, from which we determine the increase in transition temperature in tests with small-size specimens. On the basis of the results obtained we determine the conditions for the reactor vessel ensuring its safe operation.

CONTRIBUTION A L'ETUDE DE LA SURETE DES CUVES DE PRESSION DU POINT DE VUE DE LA RUPTURE.

Les auteurs étudient la sûreté des cuves de pression pour réacteurs nucléaires en établissant leur diagramme d'analyse de rupture. Celui-ci exprime les variations de la résistance à la traction, de la limite d'élasticité et de la ténacité en fonction de la température, ainsi que celles de la température d'arrêt des ruptures (TAT) en fonction des tensions. L'emploi de ce diagramme et la connaissance du champ de tensions et de la taille des défauts dans les cuves de pression permettent de déterminer la température minimale admissible pour une charge donnée, ou les paramètres (pression et température) intervenant au cours des essais de surpression et des régimes de démarrage du réacteur.

La construction d'un diagramme d'analyse de rupture est fondée sur les variations de la résistance à la traction et de la limite d'élasticité en fonction de la température, déterminées pour des échantillons normalisés et compte tenu des effets de taille évalués au cours d'expériences antérieures, ainsi que sur

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les variations de la TAT en fonction des tensions, pour des échantillons d'épaisseur normale et deux grandeurs de l'effort (2/3 de la limite d'élasticité et la limite d'élasticité elle-même). La ténacité est déterminée conformément aux recommandations de l'ASTM; en ce qui concerne les spécifications de taille minimale, les échantillons du type WOL-X sont préférables.

En plus de l'effet de taille, dont on tient compte, l'énergie accumulée et le vieillissement des matériaux (sous l'action de l'irradiation neutronique, notamment) peuvent influencer sur le diagramme d'analyse de rupture. Pour examiner le premier facteur, on a effectué des essais d'arrêt des tapures sur des tôles soumises à une charge mécanique et des cuves contenant de l'eau sous pression ou de l'eau et du gaz sous pression. Ces essais montrent que le critère de la TAT garantit une sûreté suffisante en ce qui concerne la rupture en l'absence de tapures de taille extrême.

L'influence de l'irradiation neutronique est exprimée, avant tout, par l'amélioration des propriétés de résistance, comme la limite d'élasticité et la résistance à la traction, et par la diminution de la plasticité liée à la fragilisation, caractérisée par un accroissement de la température de transition (et de la TAT). L'expérience montre que les variations de ces propriétés dépendent beaucoup de l'état initial des matériaux et de leur microstructure. On observe parallèlement une détérioration des paramètres relatifs à la résistance au choc dans la région de malléabilité.

Pour déterminer les effets de l'exposition aux rayonnements sur les propriétés des matériaux, en conditions réelles d'emploi, on fait appel au programme d'échantillons témoins, dans lequel la température de transition est déterminée au moyen d'échantillons de petite taille. L'ensemble des méthodes décrites ci-dessus permet ainsi de déterminer les régimes des cuves sous pression et de garantir la sûreté de leur exploitation.

К ВОПРОСУ О БЕЗОПАСНОСТИ КОРПУСОВ ДАВЛЕНИЯ АТОМНЫХ РЕАКТОРОВ С ТОЧКИ ЗРЕНИЯ ХРУПКОГО РАЗРУШЕНИЯ.

Для оценки безопасности корпуса давления атомного реактора используется его диаграмма прочности, т. е. зависимость прочности, предела текучести и хрупкого разрушения от температуры и зависимость температуры остановки трещины (ТОТ) от напряжения. С помощью диаграммы прочности, а также зная напряженность и величину дефектов в материале корпуса давления, можно определить наименьшую допустимую температуру при данной нагрузке или же определить параметры (давление и температуру) в ходе испытания на прочное давление и в режимах пуска во время эксплуатации.

Составление диаграммы прочности проводится на основании зависимости прочности и предела текучести от температуры, определенной на стандартных образцах, с учетом влияния масштабного фактора, согласно нашим более ранним работам, а также на основании зависимости ТОТ от напряжения на образцах действительной толщины при двух уровнях напряжений (2/3 от предела текучести и предел текучести). Хрупкое разрушение потом определяется согласно Рекомендациям ASTM; с учетом требуемых минимальных размеров выгодно использовать образцы типа WOL-X.

На диаграмму прочности, кроме учитываемого нами масштабного фактора, может также оказать влияние аккумулированная энергия, старение материала и радиационное повреждение. В целях исследований степени влияния первого фактора нами были проведены испытания на остановку трещины на плите, подвергаемой механической нагрузке, и на корпусах давления при заполнении водой или водой и воздухом под давлением. Испытания показали, что, если в материале корпуса давления не обнаруживаются трещины ненормальных размеров, критерий ТОТ обеспечивает достаточную стойкость против хрупкого разрушения.

Влияние нейтронного облучения проявляется, прежде всего, в увеличении прочностных параметров (предела текучести и прочности) и в уменьшении вязкости, связанной с охрупчиванием, которое характеризуется повышением критической температуры хрупкости и ТОТ. Изменение этих параметров сильно зависит от исходного состояния материала и от его микроструктуры, как это было показано экспериментально. Одновременно происходит уменьшение параметров ударной вязкости в вязкой области.

Для изучения влияния радиационного повреждения на свойства материала в действительных условиях используется программа образцов-свидетелей, по которой определяется повышение критической температуры при испытании на малых образцах. На основании полученных результатов определяются режимы для корпуса реактора, которые обеспечивают его безопасную работу.

CONSIDERACIONES SOBRE EL TEMA DE LA SEGURIDAD DE LA VASIJAS DE CONTENCIÓN CONTRA LA FRACTURA FRÁGIL.

Para estudiar la seguridad de la vasija de contención de un reactor nuclear se utiliza el Diagrama de Análisis de Fractura, esto es: se debe conocer la relación entre la resistencia a la tracción, el límite elástico y la resiliencia y la temperatura, y la relación entre la temperatura de detención de la rotura (CAT)

y el esfuerzo. Utilizando el Diagrama de Análisis de Fractura, y suponiendo que se conoce el campo de esfuerzos y los tamaños de las grietas, podemos determinar las temperaturas mínimas admisibles bajo una carga dada, o bien determinar los parámetros (presión y temperatura) durante la prueba de sobrepresión y durante la puesta en marcha del reactor.

La elaboración del Diagrama de Análisis de Fractura se realiza basándose en la relación entre la resistencia a la tracción y el límite elástico y la temperatura, determinado en las muestras de tamaño normalizado, considerando el efecto de tamaño de acuerdo con nuestros primeros experimentos, así como en relación del CAT con la tensión en muestras de espesor natural para dos niveles de esfuerzos ($2/3$ del límite de elasticidad y el límite de elasticidad mismo). La resiliencia se determina de acuerdo con las normas ASTM; en lo referente a los requisitos del tamaño mínimo se prefieren las muestras del tipo WOL-X.

El Diagrama de Análisis de Fractura, además del efecto de tamaño, que se considera, puede estar influido por la energía acumulada y por el envejecimiento del material, incluyendo los efectos de irradiación neutrónica. Con objeto de estudiar la primera influencia mencionada se han llevado a cabo ensayos de detención de la rotura en placas cargadas mecánicamente introducidas en vasijas de presión, ocasionada ésta por agua o agua y gas. Estos ensayos ponen de manifiesto el hecho de que el criterio de la CAT permite ofrecer suficiente seguridad de la vasija de presión contra la fractura frágil en el caso de que no existan roturas de tamaños extremos.

La influencia de la irradiación neutrónica se manifiesta, primeramente, en el aumento de las propiedades mecánicas tales como la resistencia a la tracción y el límite de elasticidad, así como en la dimensión de la plasticidad relacionada con la fragilidad, determinada por el aumento de la temperatura de transición (y la CAT). Los cambios de estas propiedades dependen muchísimo del estado inicial del material, o bien de su microestructura, como se ha descubierto experimentalmente. Al mismo tiempo se ha observado una deterioración de los parámetros relativos a la resistencia al choque en la región dúctil.

Para la investigación de los efectos de la irradiación en las propiedades de los materiales bajo condiciones reales, se utilizó un sistema de inspección de muestras, en el que se determina el aumento de la temperatura de transición mediante la ayuda de muestras de pequeño tamaño. Basándose en el método mencionado anteriormente se determinan los regímenes de la vasija de presión y se logra el funcionamiento en condiciones de seguridad.

1. ВВЕДЕНИЕ

Из практики известно, что у крупногабаритных сварных сосудов, работающих под давлением, хрупкое разрушение может произойти при их изготовлении, испытании давлением или при эксплуатации. Если первые два случая разрушения создают чаще всего только экономические затруднения, то разрушение сосуда в процессе эксплуатации может привести к катастрофическим последствиям. Поэтому вопросам безопасности корпусов атомных реакторов и их защиты от какого-либо разрушения уделяют серьезное внимание. В СССР эти вопросы систематически изучаются, начиная с 1960 года. Для этой цели на заводах "Шкода" (Пльзень) были сконструированы и построены большая испытательная машина мощностью 8000 мегаграмм (ZZ 8000) и модель корпуса реактора в натуральную величину, на которой проводились долговременные повторные испытания. С помощью машины ZZ 8000 изучалось влияние нагрузки на прочность, а также влияние температуры на остановку трещины [1,2]. Кроме того, проводились испытания частей корпуса, подвергаемых наибольшей нагрузке, в натуральную величину вплоть до разрушения (таких, как часть крышки с патрубками, часть боковых патрубков и винтовые соединения). Влияние радиационного повреждения на основной материал и отдельные сварные соединения также было основательно изучено. Результатом работы являются рекомендации по изготовлению и испытанию давлением корпуса атомного реактора А1, а также правила пускового режима реактора с точки зрения безопасности и защиты от хрупкого разрушения. Основные результаты этих работ были опубликованы раньше [1-3].

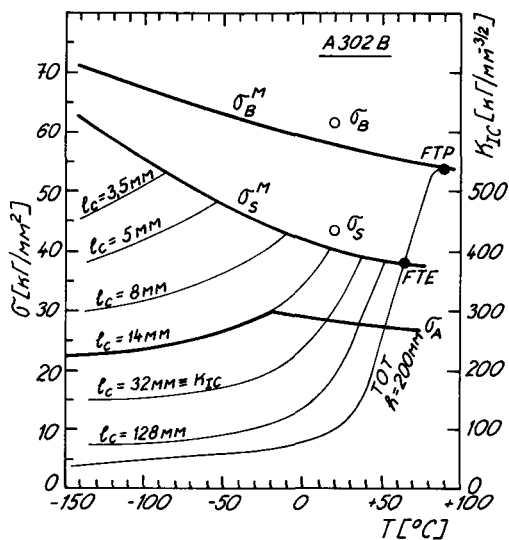


Рис.1. Диаграмма прочности стали типа А 302 В толщиной 200 мм.

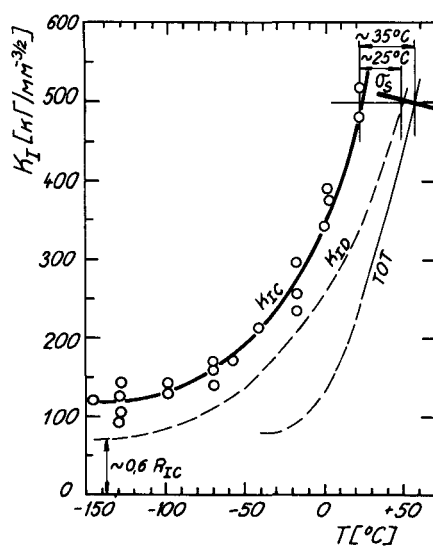


Рис.2. Сравнение зависимостей статической вязкости разрушения, динамической вязкости разрушения и температуры останова трещины (TOT).

В последние годы были достигнуты большие успехи в практическом применении механизма разрушения. В настоящее время уже представляется возможным установить более объективные критерии для обеспечения безопасности сосудов давления с точки зрения хрупкого разрушения. Поэтому мы хотим в нашей работе рассмотреть основные вопросы безопасности корпусов ядерных реакторов с точки зрения новейших достижений, а также представить последние результаты, полученные нами в результате исследования этих проблем.

2. КРИТЕРИИ БЕЗОПАСНОСТИ И ДИАГРАММА ПРОЧНОСТИ

Для обеспечения сопротивления конструкции хрупкому разрушению можно применять два метода: 1) с использованием вязкости разрушения K_{IC} или 2) с использованием температуры перехода, точнее температуры остановки трещины (ТОТ). Основу обоих методов, так же как и их взаимное соотношение, можно объяснить с помощью диаграммы прочности (см. рис.1 [2]). Диаграмма прочности показывает зависимость предела текучести, прочности и вязкости разрушения от температуры, а температуры остановки трещины – от напряжения. Кривая ТОТ делит диаграмму на две области – область пластических изломов и область хрупких изломов. В области пластических изломов разрушение происходит при пределе прочности σ_B и при образовании больших трещин. В области хрупких изломов при образовании больших трещин разрушение может произойти даже при сравнительно низком напряжении. Характеристикой сопротивления материала хрупкому разрушению является величина вязкости разрушения K_{IC} , а безопасность конструкции зависит от отношения $\sigma\sqrt{l}$ (номинальное напряжение на квадратный корень из длины трещины) к вязкости разрушения. На рис.1 представлена диаграмма прочности для стали типа А 302 В толщиной 200 мм. На диаграмме указана, кроме того, зависимость вязкости разрушения от температуры, а также полученные на ее основе величины критических длин трещин.

Статическая вязкость разрушения – K_{IC} ($\text{кг}\cdot\text{мм}^{-3/2}$) – определяется на испытательных образцах соответствующей формы, с длинной усталостной трещиной. Толщина образца и длина трещины являются лимитирующими размерами для определенного отношения K_{IC}/σ_S . Формы испытательных образцов, способы охлаждения и приложения нагрузки, метод снятия диаграммы Р-COD (усилие – раскрытие трещины) и способ ее оценки являются нормированными [4]. В результате испытаний получаем зависимость K_{IC} от температуры (см. рис.2), которая также носит переходный характер, но при низких температурах величина K_{IC} почти постоянна.

Кроме статической вязкости разрушения, в последнее время вводится понятие так называемой динамической вязкости разрушения, обозначаемой K_{ID} [$\text{кг}\cdot\text{мм}^{-3/2}$], которую можно определить на тех же образцах, что и K_{IC} , при приложении к ним нагрузки с высокой скоростью. Однако наилучшим способом определения K_{ID} является испытание клинообразного двухконсольного испытательного образца (TDCB Specimens) на специальных экспериментальных машинах, позволяющих прикладывать нагрузки с очень высокими скоростями [5]. Характер температурной зависимости у динамической вязкости разрушения K_{ID} подобен характеру такой же зависимости статической K_{IC} , однако кривая располагается ближе к низшим значениям вязкости разрушения и к высшим значениям температур (см. рис.2).

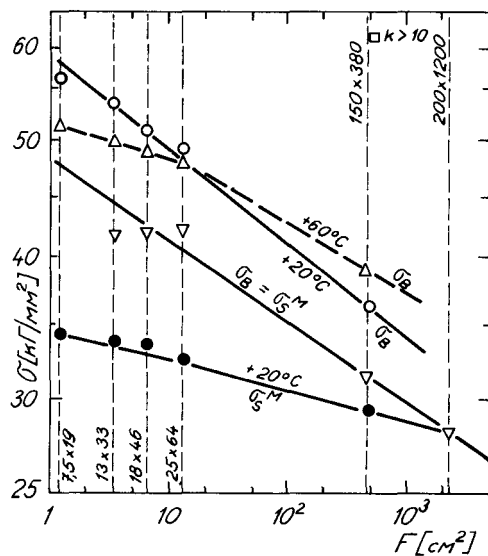


Рис.3. Зависимость прочности и предела макропластических деформаций для образцов с острым надрезом от их размера при различных температурах испытания.

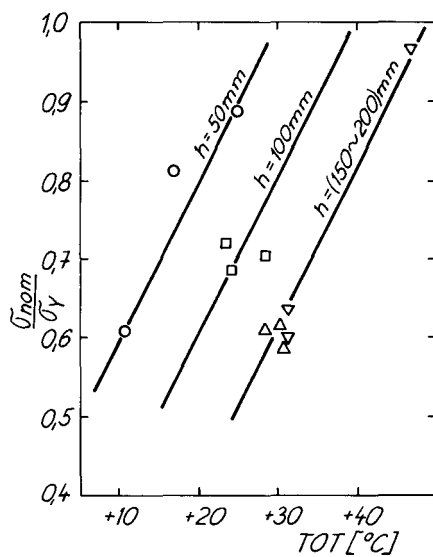


Рис.4. Зависимость температуры остановки трещины (TOT) от номинального напряжения для различных толщин образца.

Пеллини и др. [6] разработали метод испытания крупных образцов (натуральной толщины, с глубоким надрезом, имеющим в основе хрупкую наплавку) на динамический изгиб (DT Tests). В результате испытания определяется зависимость энергии разрушения E от температуры. Авторы указывают, что температура при $1/2 E_{\max}$ соответствует температуре закрытия трещины при σ_s , т.е. речь идет о точке FTE (см. рис.1).

Взаимосвязь K_{IC} , K_{ID} и TOT показана на рис. 2. Данные для K_{IC} и K_{ID} были взяты из литературы [5-7] и справедливы для стали А 533-B-1 толщиной примерно 300 мм. Зависимость TOT представлена на основе данных ΔT испытаний для того же материала. Закономерность связи, однако, носит общий характер [8]. На графиках видно, что в переходной области кривая K_{IC} по отношению к кривой K_{ID} подвинута в сторону высших значений температуры на 28°C и TOT — примерно на 35°C . Кривая TOT для материала средней толщины является, вообще, предельной кривой, так как получается при предельных скоростях деформации. Поэтому кривая TOT имеет важное значение при составлении диаграммы прочности.

Из вышеизложенного ясно, что для определения безопасности сосуда давления необходимо иметь его диаграмму прочности. Однако на отдельные характеристики диаграммы оказывают влияние многие факторы, которые необходимо учитывать. Приведем наиболее важные из них для атомных реакторов, охлаждаемых газом.

3. ВЛИЯНИЕ МАСШТАБА И ВЕЛИЧИНЫ АККУМУЛИРУЕМОЙ ЭНЕРГИИ НА ДИАГРАММУ ПРОЧНОСТИ

Нами было изучено влияние размеров образца, прежде всего его толщины, на прочностные характеристики и температуру остановки трещины. Испытывались образцы вплоть до сечения 200×1200 мм на испытательной машине ZZ 8000. Установлено [2,9], что у образцов без надреза или со средним концентратором напряжения ($K=3$) прочность не зависит от величины конструкции. При наличии острых концентраторов напряжения прочность и предел макроскопических деформаций уменьшаются в случае увеличения сечения тела и в случае хрупкого или пластического разрушения. Эта зависимость для стали типа А 212 В (13030 по ЧСН) представлена на рис.3 для температур $+20^\circ\text{C}$ и $+60^\circ\text{C}$ и для одинаковых условий разрушения $\sigma_B = \sigma_S^M$ (т.е. разрушение на уровне макропластических деформаций). Как видим, возникает линейная зависимость в логарифмических координатах как при пластическом ($+20^\circ$ и $+60^\circ\text{C}$), так и при хрупком ($\sigma_B = \sigma_S^M$) разрушениях.

Зависимость температуры остановки трещины от толщины (от 50 до 200 мм) для стали типа А 212 В (13030 по ЧСН) показана на рис.4. Зависимость TOT в области напряжения $\sigma_{\text{ном}} / \sigma_S = (0,5 \div 1,0)$ — приблизительно линейная. С повышением $\sigma_{\text{ном}}$ от $0,5 \sigma_S$ до $1,0 \sigma_S$ TOT увеличивается на $22 \div 24^\circ\text{C}$. При увеличении толщины образца от 50 до 100 мм и от 100 до 150 мм TOT повышается в пределах $7 \div 9^\circ\text{C}/50$ мм. Дальнейшее увеличение толщины от 150 до 200 мм практически уже не оказывает влияния на TOT.

В последнее время нами изучалось также влияние высокой аккумулярованной энергии, которая возникает, например, в сосудах давления, наполненных газом, на механизм хрупкого разрушения. Это влияние исследовалось, прежде всего, с точки зрения критерия TOT [10]. Испыты-

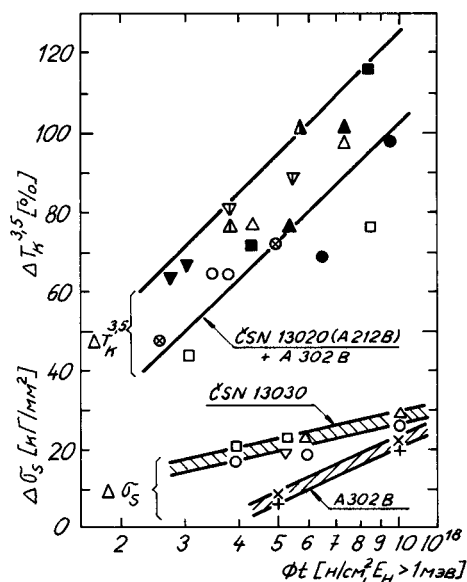


Рис.5. Зависимость роста предела текучести и критической температуры перехода от нейтронной дозы для стали типа А 212 В (13 030 по ЧСН) ($\sigma_{S \text{ необл}} \approx 25 \text{ кг} \cdot \text{мм}^{-1/2}$) и типа А 302 В ($\sigma_{S \text{ необл}} = 33\text{--}41 \text{ кг} \cdot \text{мм}^{-1/2}$).

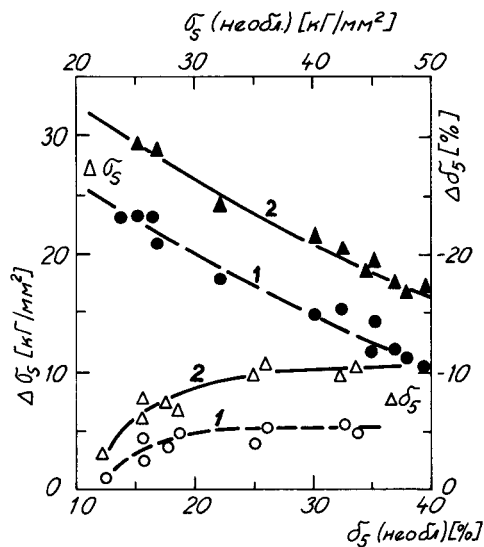


Рис.6. Зависимость роста предела текучести и снижения общего удлинения от их исходных значений для стали типа А 212 В (13 030 по ЧСН) после облучения при температуре примерно 80°C :

- 1 - $6 \cdot 10^{18}$ нейтр / см^2 ,
- 2 - $1 \cdot 10^{19}$ нейтр / см^2 .

вались сосуды диаметром 1200 мм, с толщиной стен 30 мм, при давлении $p = 90 \text{ ат}$, нагружаемые жидкостью и воздухом (20% воздуха, 80% воды). Вдоль оси сосудов был создан температурный градиент. Производилось сравнение ТОТ, определенной при приложении механической нагрузки к испытательной плите, с ТОТ, полученной при испытании сосудов с водой или с водой и воздухом. Эксперименты показали, что до тех пор, пока не возникнет слишком большая трещина, которая будет уже недопустимой с точки зрения плотности сосуда, воздушная среда не будет влиять на ТОТ материала, т.е. ТОТ для всех трех вышеуказанных случаев практически одинакова. Последнее могло бы означать, что вопреки выводам Никольза и др. [11] критерий ТОТ мог бы применяться и для сосудов с газовым наполнением. Указанные выше работы еще не окончены, эксперименты будут продолжаться.

4. ВЛИЯНИЕ ИЗМЕНЕНИЙ МЕХАНИЧЕСКИХ СВОЙСТВ МАТЕРИАЛА В РЕЗУЛЬТАТЕ НЕЙТРОННОГО ОБЛУЧЕНИЯ НА ДИАГРАММУ ПРОЧНОСТИ

Нейтронное облучение производит изменение как в структуре материала, так и в его свойствах, из которых наибольшее значение имеют: увеличение предела текучести (т.е. практическое упрочнение материала), уменьшение пластических свойств (удлинения и сужения) и значительное повышение критической температуры перехода. Эти изменения, однако, в существенной мере зависят как от нейтронной дозы, так и от температуры облучения и, наконец, от самого материала, т.е. от его исходного состояния. Так как два первых вопроса были уже достаточно хорошо изучены, остановимся на третьем. Исходное состояние материала оказывает сильное влияние на изменение механических свойств в процессе облучения [12,13], независимо от того, какими способами это состояние было достигнуто (различными термическими обработками, изменением технологии и т.д.). У малоуглеродистых и низколегированных сталей (типа А 212 В и А 302 В по АСТМ) после термообработки нормализацией и отпуском влияние исходного состояния материала проявляется, прежде всего, на его свойствах при растяжении. Изменение же критической температуры перехода не зависит от исходного состояния материала (см. рис. 5 и 6).

Рис. 5, а еще лучше рис. 6, характеризует влияние исходного состояния материала на его свойства при растяжении – предел текучести и удлинение, а также на изменение этих свойств в процессе облучения (чем больше предел текучести материала перед облучением, тем меньше его увеличение; чем больше удлинение материала перед облучением, тем больше его уменьшение). Однако для материалов с удлинением больше чем на 20% уменьшение удлинения потом не зависит от начальной величины.

Сильное влияние оказывает, прежде всего, термообработка материала, особенно в тех случаях, когда достигнутые микроструктуры не являются такими же стабильными, как после нормализации и отпуска. Результаты ряда экспериментов для стали типа А 302 В при различных условиях термообработки, представленные на рис. 7 [14], показали, что у сталей с наилучшей радиационной устойчивостью не достигается такая хорошая комбинация прочностных характеристик и вязкости, которая требуется для практического использования корпусов. Наиболее подходящая комбинация прочностных свойств и вязкости как в начальной стадии до облучения,

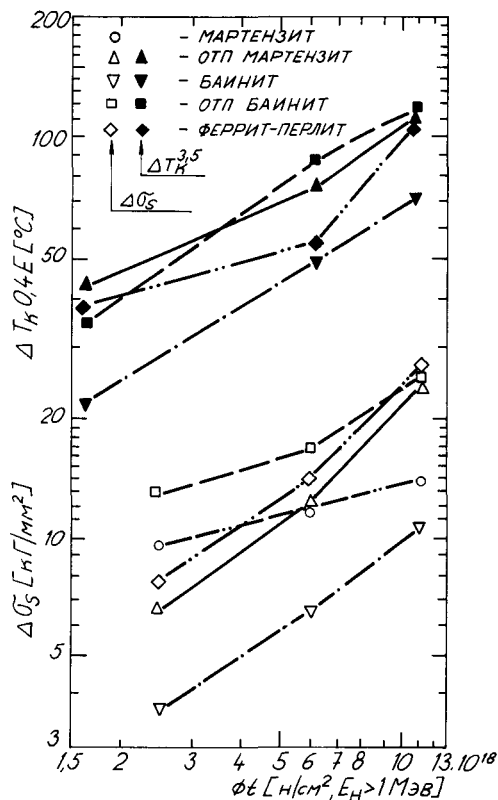


Рис.7. Зависимость прироста предела текучести и критической температуры перехода от нейтронной дозы для стали типа А 302 В при разных условиях температурной обработки.

так и в стадии после нейтронного облучения (при температуре примерно 65°C) была найдена у микроструктуры, полученной быстрым охлаждением с аустенизационной температурой и последующим высоким отпуском.

Для обеспечения безопасной работы корпуса под давлением наиболее важным является, прежде всего, сдвиг критической температуры перехода, т.е. связанные с этим изменения вязкости материала. В условиях облучения правильное установление обоих изменений становится серьезной проблемой. Чаще всего применяются два типа испытаний – на статический изгиб или растяжение (образцы типа WOL) и на ударный изгиб (образцы типа Шарпи-V), – разница между которыми заключается, прежде всего, в скорости приложения нагрузки. Разность скоростей составляет несколько порядков. Если испытания на ударный изгиб позволяют обычно определить только изменение критической температуры перехода (а также уменьшение ударной вязкости), то статические испытания дают возможность определить как величину вязкости разрушения K_{IC} , так и величину раскрытия трещины COD. Сравнение результатов испытаний на статический и ударный изгибы, проведенных нами со сталью типа 13 030 по ЧСН (тип А 212 В) после облучения при температуре примерно 175°C , представлено на рис.8.

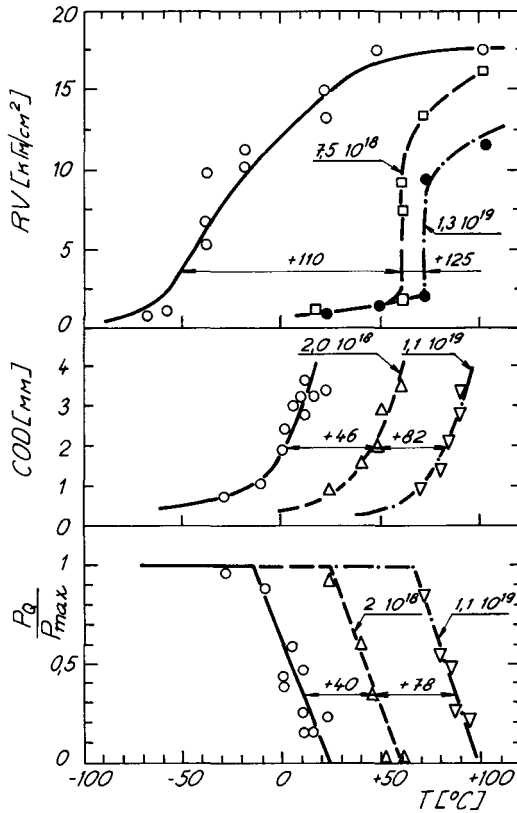


Рис.8. Зависимость сдвигов критических температур перехода от нейтронной дозы для испытания на статический изгиб (а,в) и для испытания ударной вязкости (с) стали типа А 212 В (13 030 по ЧСН).

При испытаниях на статический изгиб определялась относительная величина скачка в хрупкой области [15] (по отношению к величине полного разрушения, т.е. P_0/P_{max}) и величина COD. На графиках видим, что обе величины показывают приблизительно подобный сдвиг критической температуры перехода. При испытаниях на ударный изгиб величина этого сдвига немного больше, что объясняется, скорее всего, более высокой скоростью приложения нагрузки. Эти результаты полностью совпадают с результатами работы [16]. Важным является также то обстоятельство, что пороговая величина вязкости разрушения K_{IC} (в нашем случае данные определялись при -196°C) после облучения уменьшилась (так, от начальной величины примерно в $125 \text{ кГ}\cdot\text{мм}^{-3/2}$ после облучения дозой $1 \cdot 10^{19}$ быстр.нейтр./см² она уменьшилась на $\sim 80 \text{ кГ}\cdot\text{мм}^{-3/2}$). Это свидетельствует о понижении кривой температурной зависимости вязкости разрушения, которое оказывает неблагоприятное влияние на безопасность корпуса.

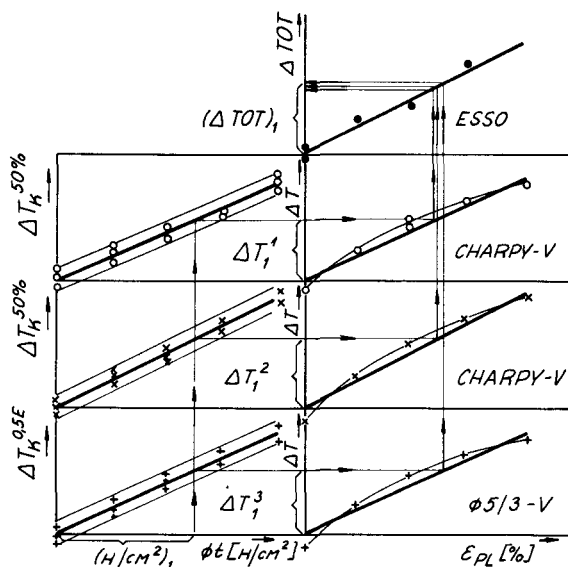


Рис.9. Схема коррекции прироста температуры остановки трещины (TOT) на прирост критических температур, определенных на малых образцах после облучения и искусственного старения.

5. КОРРЕКЦИЯ ДИАГРАММЫ ПРОЧНОСТИ В ХОДЕ ЭКСПЛУАТАЦИИ КОРПУСА РЕАКТОРА

В разделе 4 нами было указано, как под влиянием нейтронного облучения изменяются свойства стали. Эти изменения необходимо учитывать в диаграмме прочности. Наибольшее значение имеет повышение величины предела текучести и, прежде всего, повышение критической температуры перехода или TOT. Между температурами перехода, с точки зрения различных критериев, не существует достаточно точной и общепринятой корреляции. В противоположность этому, изменения критических температур перехода под влиянием нейтронного облучения (ΔT_K) по различным критериям довольно хорошо совпадают [17]. Практически величину сдвига критической температуры перехода можно определить только с помощью малых испытательных образцов. Для того чтобы исключить влияние метода определения ΔT_K на величину ΔTOT , пользуемся схемой, показанной на рис.9 [18]. На испытательных образцах натурной толщины нами было установлено изменение температуры остановки трещины после искусственного старения различной пластической деформацией.

На меньших образцах, вырезанных из больших конструкций, определялось ΔT_K по критериям, которые применялись на образцах-свидетелях. Затем, сравнением отдельных ΔT_K получим истинное значение изменения ΔTOT . После определения ΔTOT и $\Delta \delta_s$ можно построить диаграмму прочности для данного состояния сосуда давления. Режимы пуска в эксплуатацию и эксплуатации реактора должны быть приведены в соответствие с этой диаграммой прочности, с тем чтобы не произошло возможного хрупкого разрушения.

6. ЗАКЛЮЧЕНИЕ

В работе было указано, что анализ безопасности корпуса атомного реактора с точки зрения его сопротивления хрупкому разрушению можно проводить с помощью диаграммы прочности. В ходе эксплуатации корпуса атомного реактора изменяются механические свойства материала, а также может произойти увеличение дефектов в стенках корпуса. Все эти изменения оказывают значительное влияние на его хрупкое разрушение. Взаимная зависимость основных характеристик сопротивления материала хрупкому разрушению K_{IC} , K_{ID} и TOT , приведенная в разделе 2, дает возможность провести более глубокий и точный анализ, и поэтому она представляет ценность для достижения более надежной безопасности и более экономичных условий эксплуатации атомных реакторов.

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REVIEW OF SAFETY ASPECTS OF NUCLEAR POWER PLANTS IN JAPAN

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Abstract-Résumé-Аннотация-Resumen

REVIEW OF SAFETY ASPECTS OF NUCLEAR POWER PLANTS IN JAPAN

Studies on the safety aspects of nuclear power plants in Japan are summarized. Concerning the safety of nuclear structural components, (i) stresses in nuclear vessel and shell intersections were analysed by various methods, theoretically and experimentally, (ii) the basic properties of low-cycle fatigue strength of nuclear vessel steels were investigated, (iii) brittle fracture evaluation tests were performed to clarify the effects of heat treatment and irradiation embrittlement on nuclear vessel steels, and (iv) studies on the non-destructive inspection of weld defects and metallurgical features of the stainless-steel weld overlay were carried out.

Concerning the safety of the JPDR pressure vessel, the following studies were carried out (i) remote non-destructive inspections of the inner walls including the nozzle corners; (ii) low-cycle fatigue studies using model vessels and laboratory test pieces, (iii) corrosion-fatigue tests of the vessel steel in a high-temperature aqueous environment, and (iv) the monitoring of changes in NDTT and Charpy-V upper shelf energy of the vessel steel.

Two co-operative studies conducted by the Nuclear Safety Research Association concerning fuel claddings and fission-product release in light-water-cooled power reactors are summarized. Mechanical properties of fuel claddings under coolant-loss accident conditions, and the effect of various defects and a directionality change of hydrides in Zircaloy due to thermal cycling were investigated. In the fission-product release studies, formation mechanisms and the behaviour of CH_3I in nuclear fuels, in high-temperature water and in reactor containment were studied.

The basis of judgment on the probabilistic safety evaluation of nuclear facilities is proposed using environmental risk analysis. Firstly, the "base-risk" (minimum environmental risk) is found. Secondly, the "safety evaluation standard risk" is determined, which allows for future lessening of the individual risk. And finally, taking various factors into consideration, the "reference risks" are proposed both for accident conditions and for normal operation.

ETUDE DES ASPECTS DES CENTRALES NUCLEAIRES JAPONAISES RELATIFS A LA SURETE.

Les auteurs rendent compte d'études faites sur les aspects des centrales nucléaires japonaises relatifs à la sûreté. En ce qui concerne la sûreté des composants structuraux, 1) on a étudié les efforts aux intersections tuyère-cylindre de la cuve du réacteur par diverses méthodes sur les plans théorique et expérimental, 2) on a fait des travaux de recherche sur les propriétés fondamentales de la résistance à la fatigue des aciers pour cuve de pression sous l'effet d'efforts cycliques faibles, 3) on a fait des essais d'évaluation de la rupture par fragilité pour préciser les effets du traitement thermique et de la radiofragilisation sur les aciers pour cuve de pression, et 4) on a fait des études sur l'inspection non destructive des défauts des soudures et sur les caractéristiques métallurgiques du revêtement en acier inoxydable des soudures.

En ce qui concerne la sûreté de la cuve de pression du réacteur de puissance japonais de démonstration (JPDR), on a fait les études suivantes 1) inspections non destructives à distance des parois internes, y compris les angles de la tuyère; 2) études de fatigue sous l'effet d'efforts cycliques faibles sur des cuves prototypes et des éprouvettes; 3) essais de fatigue et de corrosion des aciers pour la cuve de pression dans une atmosphère humide à haute température, et 4) observation des variations de la température de

transition (ductilité zéro) des aciers et de l'énergie correspondant au palier supérieur dans l'essai des aciers à la machine de Charpy.

Les auteurs rendent compte également de deux études faites en coopération avec la Nuclear Safety Research Association sur le gainage du combustible et le dégagement de produits de fission dans les réacteurs de puissance refroidis à l'eau légère. Des travaux de recherche ont été faits sur les propriétés mécaniques des gaines dans les conditions d'un accident par perte de caloporteur ainsi que sur les effets de divers défauts et d'une réorientation des hydrures dans le Zircaloy dus au cycle thermique. Dans les études sur le dégagement des produits de fission, on a accordé une attention particulière aux mécanismes de formation de CH_3I dans les combustibles nucléaires, l'eau à haute température et l'enveloppe de sécurité du réacteur, ainsi que sur le comportement de ce composé.

Les auteurs proposent un critère pour l'évaluation probabiliste de la sûreté des installations nucléaires en se fondant sur l'analyse des risques pour l'environnement. En premier lieu, ils définissent un « risque de base » (risque minimal pour l'environnement). En deuxième lieu, ils déterminent le « risque type pour l'évaluation de la sûreté » qui permettra de réduire le risque individuel. Enfin, en prenant en considération divers facteurs, ils proposent des « risques de référence » pour le cas d'accident et pour les conditions de fonctionnement normal.

ОБЗОР ПО ВОПРОСАМ БЕЗОПАСНОСТИ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ В ЯПОНИИ.

Суммированы исследования по вопросам безопасности атомных электростанций в Японии. Что касается безопасности конструктивных элементов ядерных реакторов, то (i) различными методами, теоретически и экспериментально, были изучены напряжения в местах соединения различных узлов с реакторным корпусом и оболочкой, (ii) были исследованы основные характеристики малоциклового усталостной прочности сталей реакторных корпусов, (iii) были проведены испытания по оценке хрупкого разрушения для выяснения влияния тепловой обработки и радиационного охрупчивания на стали реакторного корпуса и (iv) были проведены проверки без разрушения дефектов сварных швов и металлургических характеристик наплавов из нержавеющей стали.

Для определения безопасности корпуса давления реактора JPDR были проведены следующие исследования: (i) дистанционный контроль без разрушения внутренних стенок, включая углы патрубка; (ii) исследования малоциклового усталости с использованием модели корпуса и образцов для лабораторного испытания; (iii) испытания на коррозионную усталость стали корпуса в воде при высокой температуре и (iv) проверка изменений переходной температуры нулевой пластичности и верхней пороговой энергии стали корпуса, определенной на образцах Шарпи (с V-образным вырезом).

Обобщены два совместных исследования топливных покрытий и утечки продуктов деления в энергетических реакторах с водяным охлаждением, проведенные Ассоциацией исследований по ядерной безопасности. Изучены механические характеристики топливных покрытий в аварийных условиях при потере теплоносителя, а также влияние различных дефектов изменения ориентированности гидридов в циркалоев вследствие тепловых колебаний. При исследованиях утечки продуктов деления были изучены механизмы образования и поведение CH_3I в ядерных видах топлива, в горячей воде и в противоаварийной оболочке реактора.

В основу осуществления вероятной оценки безопасности ядерных установок предлагается положить анализ опасности для окружающей среды. Во-первых, найдена "базовая опасность" (минимальная опасность для окружающей среды). Во-вторых, определен "стандартный риск при оценке безопасности", который в будущем позволит снизить индивидуальную опасность. И, наконец, принимая во внимание различные факторы, предлагаются "эталонные опасности", как для аварийных условий, так и для нормальной эксплуатации.

RESUMEN DE LOS ESTUDIOS SOBRE LA SEGURIDAD DE LAS CENTRALES NUCLEARES EN EL JAPÓN.

Se resumen los estudios sobre las cuestiones de seguridad de las centrales nucleares en el Japón. Con referencia a la seguridad de los componentes estructurales nucleares a) se han analizado teórica y experimentalmente por diversos métodos las tensiones en las vasijas nucleares y en las conexiones de los casquillos; b) se han investigado las propiedades básicas de resistencia a la fatiga a bajo número de ciclos de los aceros de las vasijas nucleares; c) se han realizado pruebas de valoración de fractura por fragilidad para aclarar los efectos del tratamiento térmico y de la fragilidad por irradiación en los aceros de las vasijas nucleares, y d) se han realizado estudios sobre la inspección no destructiva de los defectos de soldadura y de las características metalúrgicas del recubrimiento por soldadura de acero inoxidable.

Con referencia a la seguridad de la vasija de presión del JPDR (reactor de potencia japonés de demostración) se han realizado los siguientes estudios: a) inspecciones a distancia no destructivas de las paredes interiores, incluyendo los codos de las toberas, b) estudios de fatiga a bajo régimen de ciclos

utilizando vasijas modelo y elementos de laboratorio; c) ensayos de fatiga por corrosión del acero de la vasija en un ambiente acuoso a temperatura elevada, y d) vigilancia de las variaciones de la temperatura de transición a la ductilidad nula y de la energía en el ensayo Charpy con entalla en V, del acero de la vasija.

Se resumen dos estudios cooperativos dirigidos por la Asociación de investigación de seguridad nuclear, relativos a las vainas del combustible y a los escapes de productos de fisión en los reactores de potencia refrigerados con agua ordinaria. Se han investigado las propiedades mecánicas de las vainas del combustible en condiciones de accidente por pérdida de refrigerante, así como los efectos de varios defectos y una variación de resistencia direccional de los hidruros en el circaloy debida al ciclado térmico. En los estudios sobre escapes de productos de fisión, se han examinado los mecanismos de formación y el comportamiento del CH_3I en los combustibles nucleares, con agua a alta temperatura y en la contención del reactor.

Se exponen los fundamentos de la valoración probabilística de la seguridad de las instalaciones nucleares empleando el análisis de riesgos ambientales. En primer lugar se encuentra el « riesgo base » (riesgo mínimo ambiental). En segundo lugar se determina el « riesgo estándar para valoración de seguridad » que permite en el futuro aminorar el riesgo individual. Por último, tomando en consideración varios factores se proponen los « riesgos de referencia » para condiciones de accidente y para funcionamiento normal.

1. STUDIES ON SAFETY OF NUCLEAR STRUCTURAL COMPONENTS

1.1. Experimental and theoretical stress analysis

The stress distribution in the drywell of a containment vessel was analysed by the finite difference method, where Sepetoshi's method was extended and spacings of the mesh were suitably selected for the particular locations of the vessel [1].

Based on the shell theory, the stress analysis of the nozzle-to-cylinder intersection of a reactor pressure vessel subjected to internal pressure was performed; Green-Zerna's equation was applied to the cylinder part and Flügge's equation was applied to the nozzle part. Singular solutions for both equations were introduced and a compatibility condition was satisfied on the intersecting line of the cylinder and the nozzle by the principle of complementary energy minimum [2].

A comparison of the analytical values derived from the two methods with experimental ones revealed good agreement.

Applying the finite-element method, the effects of intersecting angles on stress distributions of obliquely-intersected pipes subjected to internal pressure or in-plane force were investigated. In the analysis, Hellinger-Reissner's variational method was applied and the curved shell was approximated by a network of flat plate elements. The mesh pattern of the shell is illustrated in Fig. 1; the circumferential and longitudinal stress distributions in the longitudinal plane are shown in Fig. 2 [3].

A three-dimensional photoelastic method is widely used for experimental stress analysis; Fig. 3 illustrates an example of the multinozzle vessel analysed [4].

1.2. Fatigue strength properties of model vessels and materials

A series of experimental studies on the law of cumulative damage, notch effect, ratchet effect, and so on were performed to investigate the low-cycle fatigue and the thermal fatigue strength properties of nuclear vessel steels and their welded joints [5].

Further, regarding the notch effect, strain-controlled and stress-controlled, low-cycle fatigue tests were conducted together with strain analysis at the notch tip. It was clarified that the fatigue life for crack initiation was a function of the peak value of effective strain range at the notch tip, while the crack propagation was nearly independent of the notch [6].

The fractured surfaces of Steam Raising Unit (SRU) tubes in a gas-cooled nuclear power plant were compared with the reference fractured surfaces obtained with fatigue tests on full-size tube specimens of the same material by utilizing microfractographic and X-ray microbeam techniques. The fractured surface shown in Fig. 4 was taken from a fatigue specimen which failed at 3960 cycles of alternating axial loading [7].

Various experiments on the fatigue strength of a nozzle part under pulsating internal pressure were conducted; results on the large model vessels of the Japan Power Demonstration Reactor (JPDR) are described in Section 2.

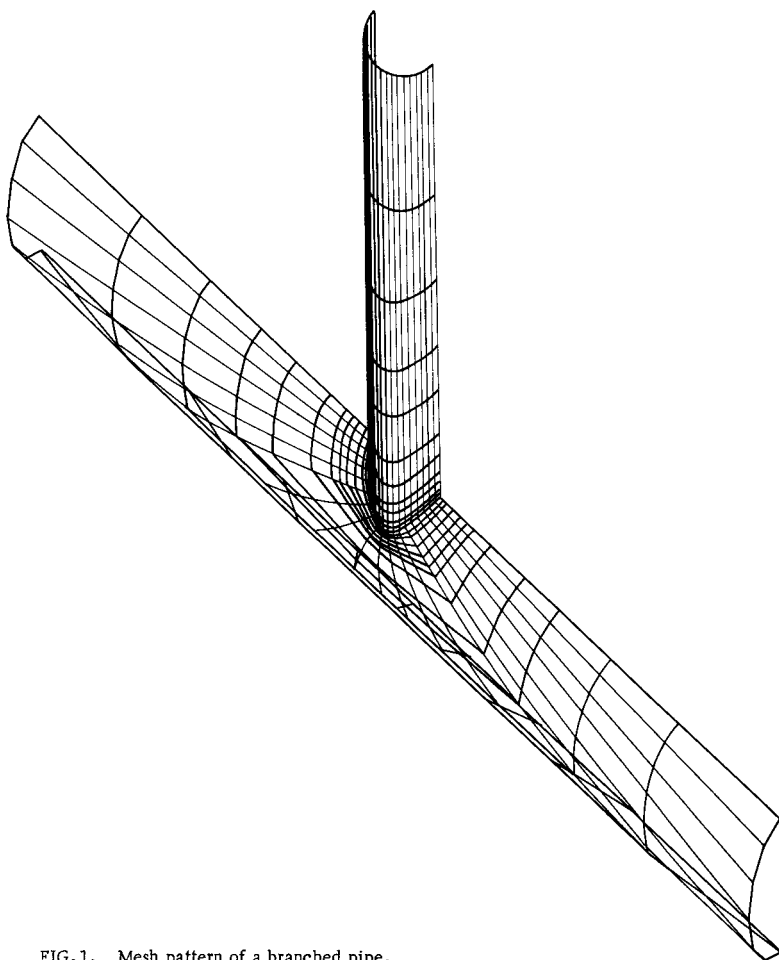


FIG. 1. Mesh pattern of a branched pipe.

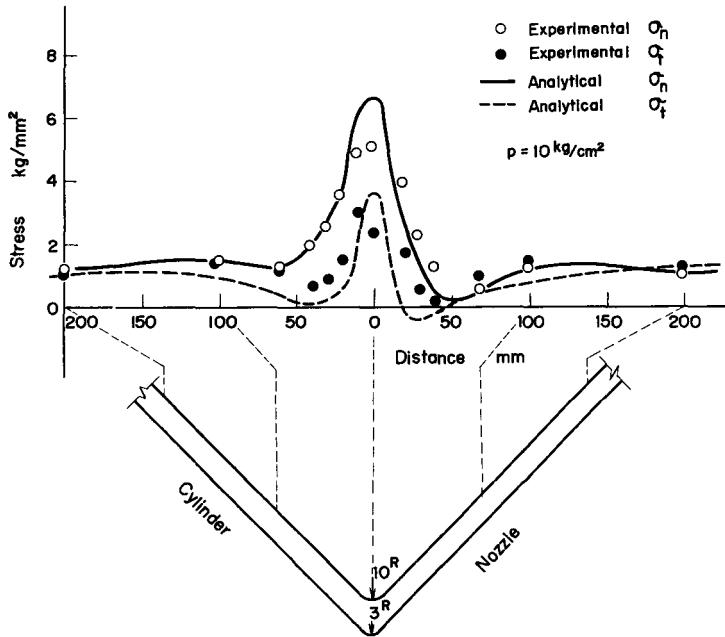


FIG.2. Outer-surface stress distributions of a pipe.

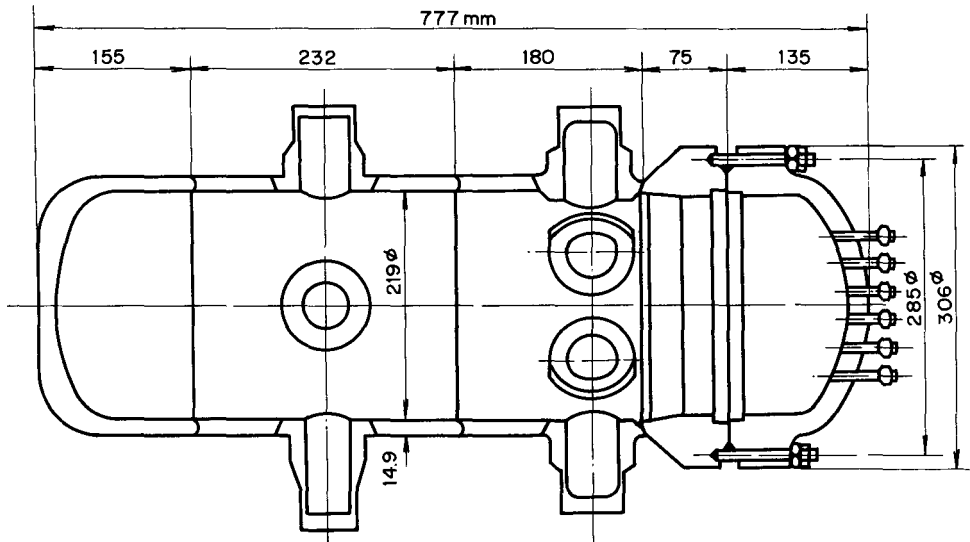


FIG.3. Photoelastic model vessel.



FIG. 4. Microfractograph of fracture surface of fatigued specimen. No. of cycles to fracture: 3960.

1.3. Brittle fracture of base metals and welded joints

To investigate the effects of thermal history in fabrication on the brittle fracture properties of nuclear vessel steel and its welds, various brittle fracture evaluation tests were performed on materials and their welded joints subjected to various kinds of heat treatments. Figure 5 illustrates the relation between half-length of the initial crack and brittle fracture initiation temperature for an infinite plate obtained from a deep-notch tension test under the assumption that uniform tensile stress is equal to $0.4 \sigma_Y$ [8].

A series of experiments were conducted to determine a suitable heat treatment condition for obtaining the desired mechanical properties of nuclear vessel steels [9].

Fracture toughness evaluation tests were performed on various types of steels to evaluate potential irradiation embrittlement, and a close correlation between the test results was found [10].

The brittle fracture properties of thick plates for nuclear vessels were investigated by using a 4000-ton tensile testing machine [11].

Studies were made on the development of a programmed test method which was capable of verifying the mechanical properties of thick plates for nuclear vessels, where small specimens simulating the cooling curves from the austenitized temperature of vessel segments were used [12, 13].

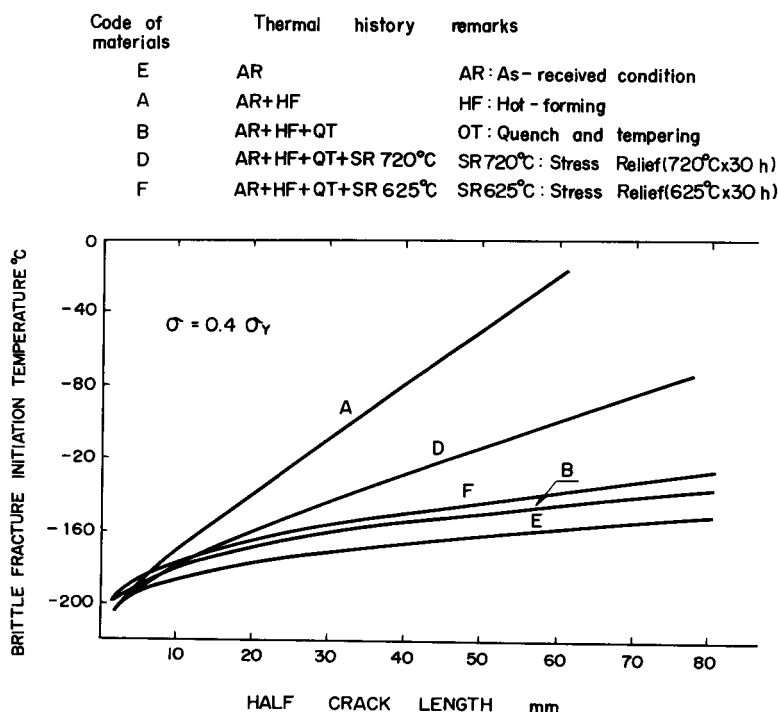


FIG. 5. Brittle fracture initiation temperature versus half crack length in infinite plate.

1.4. Inspection

Research was conducted on the correlation between actual weld defects and the non-destructive inspection results of nuclear vessel steels and their welded joints, and on the influence of weld defects on the strength of welded joints [14].

To prevent defects of the stainless-steel weld overlay, metallurgical studies were made and performance tests such as corrosion tests and mechanical property tests were conducted [15].

A full-size model of a nuclear pressure vessel was fabricated and tested to investigate the mechanical and chemical properties of its components, and to determine the fabrication and inspection practices to be used in manufacturing them [16].

The electric resistance probe method for in-service inspection of nuclear pressure vessels is described in Section 2.

2. STUDIES ON SAFETY EVALUATION OF THE JPDR PRESSURE VESSEL

2.1. Post-operation inspection of the JPDR pressure vessel

Since its initial criticality on 22 August, 1963, the JPDR has continued in operation. During periodic plant inspections in 1966, a number of fine cracks were detected on the hand-welded cladding surface of the top head of the pressure vessel. This discovery of cladding cracks emphasized the necessity for inspecting the inner surface of the vessel.

After the plant shut-down on 11 May, 1968, Suguri et al. [17] inspected the pressure vessel, using non-destructive techniques such as using a borescope, the resistance-probe system, ultrasonic testing apparatus, and the dye-penetrant method.

As a result of the borescopic inspection, a few cracks were observed on the shell side area of the west forced-circulation outlet nozzle around the 3:00 o'clock position. From the analysis of data measured by the resistance-probe system, several indications were also obtained on the same area, and the depths of these cracks were estimated to be up to about 3 mm, whereas the thickness of the overlaid cladding was 6.5 mm. An indication observed on the same area by ultrasonic inspection was estimated to be equivalent to that from the 20 mm long, 1 mm deep, and 0.5 mm wide notch.

Those described above were the most notable flaws observed by the three methods. The inspection results on flaws other than the above are summarized below:

(1) No cracks were detected on the area around the east forced-circulation outlet nozzle.

(2) It was considered that a crack-like flaw possibly existed on the shell side of the west forced-circulation outlet nozzle around the 10:30 o'clock position. Suspicious indications were detected by the borescope and the resistance-probe system in this position, although nothing was detected by ultrasonic testing.

(3) No cracks were detected on the area around the east and west forced-circulation inlet nozzles.

(4) Some indications were obtained by the borescope and/or by resistance-probe inspection on the inner surface of the bottom head. Because these indications were suspicious but not clear, it was considered that cracks possibly existed in these areas.

(5) As a result of dye-penetrant inspection, several indications were detected on the inner surface of the upper part of the vessel shell. The resistance-probe method was applied to examine cracks detected by dye-penetrant inspection. Maximum depth of the cracks was 2 mm.

It should be noted that these indications were concentrated on the hand-welded cladding surface, as was the case with the vessel top head.

From the measurements of the depth of each indication estimated by the resistance-probe system, it was concluded that there was no defect reaching the interface between the base metal and the cladding material in the pressure vessel.

The second inspections were conducted in the 1969-1970 period by using similar techniques, and the results were almost similar to those of the first inspections.

2.2. Low-cycle fatigue testing on model steel vessels

Because the fatigue life of an actual pressure vessel depends on many different factors compared with that of small-size laboratory specimens, Fukimura et al. [18] surveyed the crack propagation through the base metal of the pressure vessel models, which contained notched flaws in the cladding at locations where the highest stress concentration existed.

The results of these tests are shown in Fig. 6. Testing pressures were 120 kg/cm² for model No. 1 and 150 kg/cm² for No. 2. These pressures

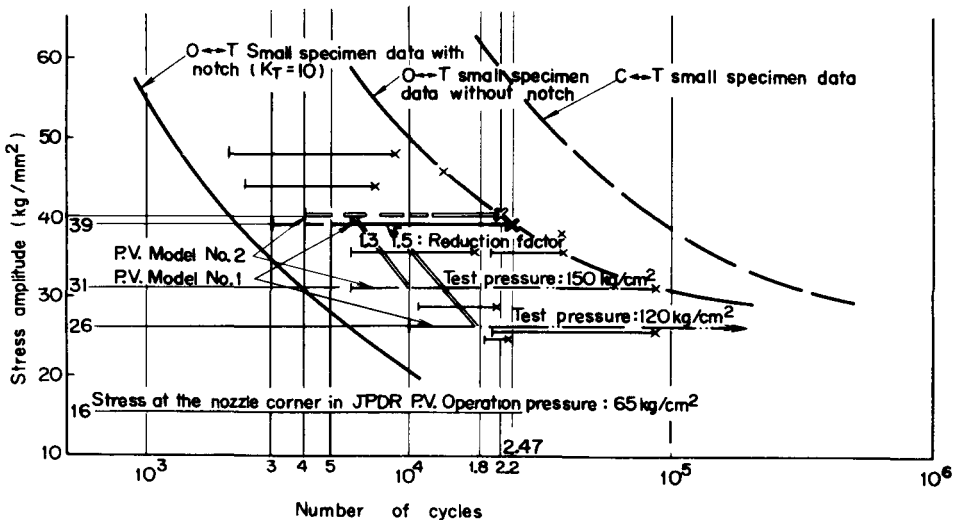


FIG. 6. Results of cyclic pressure loading test on vessel models with notched nozzle.

gave stress values of 51.4 kg/mm^2 for model No. 1 and 62 kg/mm^2 for No. 2 at the vertical, inside corner of the nozzle attached to the model vessels under a notch-free condition. The highest stress value estimated for the JPDR pressure vessel is 32.5 kg/mm^2 . The theoretical stress concentration factor for the notched flaws in the models fell within 8 - 10.

In regard to the notched nozzles, shortened life was obtained with about 3000 to 5000 cycles for crack initiation and about 20 000 cycles for final fracture. Such reduction is due to the stress concentration effect, and the effective factor is estimated to be approximately 1.5, which is smaller than the value 2.4 obtained by laboratory testing with small specimens.

2.3. Low-cycle fatigue of A302B steel in high-temperature simulated reactor environment

Through a series of corrosion fatigue studies at JAERI, a new view is being obtained on the effect of environment on the low-cycle fatigue of steels used for pressure vessels. Kondo et al. [19] developed a technique for studying fatigue crack propagation in a high-temperature pressurized-water environment.

Their work covers the effect of certain variables on the fatigue life and crack propagation character of the material; the most important of these are temperature, stress or strain amplitude, frequency of loading, and water chemistry. The fatigue life of the steel was found to be generally reduced in relation to the ambient air environment. The effect was largely dependent on any one of the above-listed parameters in a characteristic manner, as follows. Elevating the test temperature to around 200°C caused a systematic increase in the fatigue crack growth rate, while the trend was sharply reversed at higher temperatures, and a comparatively moderate effect was produced under normal reactor service conditions. In an aqueous environment, the crack growth rate was more sharply dependent on the strain or stress range as compared to ambient conditions. A fact particularly worth noting is that the crack growth is effectively interrupted by local corrosion action at the tips of the growing cracks as the loading frequency is lowered to around 3 to 1 cycles/min; below this, incubation became increasingly frequent and long-lasting (typically illustrated in Fig. 7). This behaviour was found to have a close association with the water chemistry of the environment.

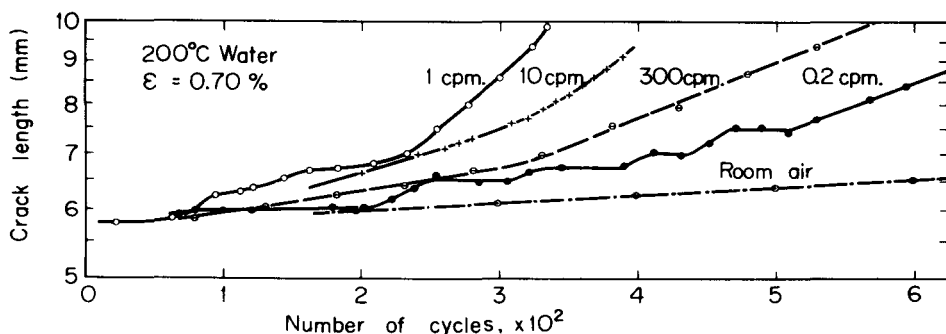


FIG. 7. Fatigue crack propagation in ASTM A302B steel in a high temperature aqueous environment.

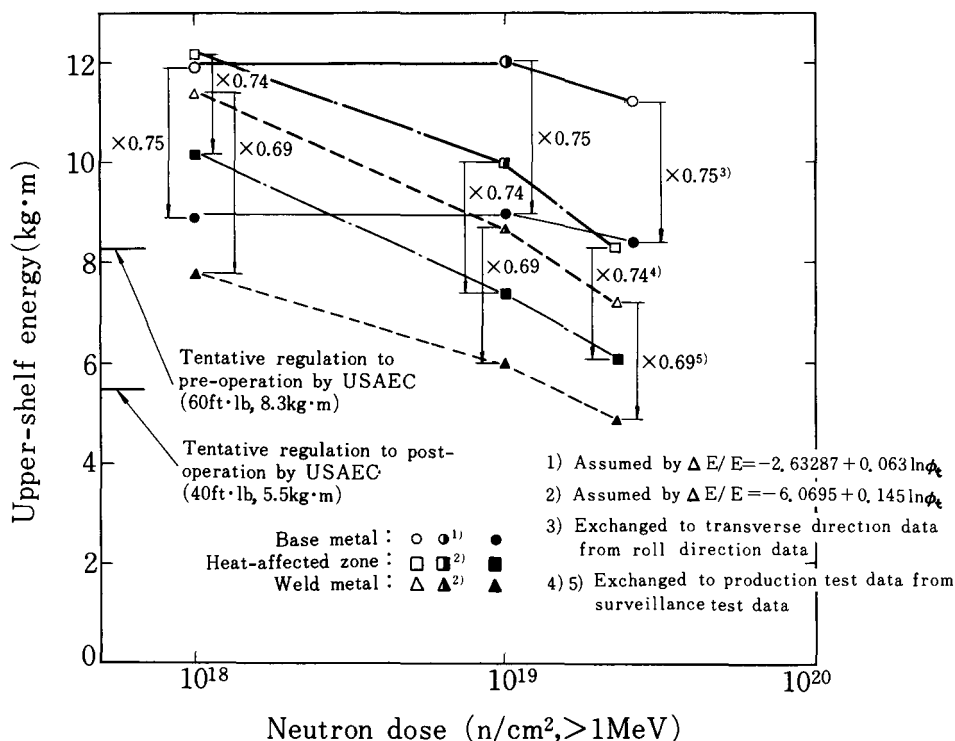


FIG.8. Results of surveillance tests - Reduction of upper-shelf energy on Charpy-V test.

2.4. Irradiation embrittlement of pressure vessel steel

The initial nil ductility transition (NDT) temperature of the vessel wall near the reactor core was about 20°C; the results of surveillance tests indicate that in the assumed reactor life of 20 full power years, the NDT temperature will reach about 120°C, provided that ΔTr_{30} equals 100 degC for a fast-neutron fluence of $2 \times 10^{19} n/cm^2$ ($E > 1 \text{ MeV}$).¹

In addition to a shift of the transition temperature, a reduction of upper-shelf energy due to irradiation as shown in Fig.8 should be noted.

From the results of the surveillance test, Kawasaki et al. [20] concluded the following:

(1) Shift of the NDT temperature of the JPDR pressure vessel steel due to irradiation will be about 100 degC, after irradiation to $2 \times 10^{19} n/cm^2$ ($E > 1 \text{ MeV}$).

(2) The reduction of upper-shelf energy is sufficiently small for the base metal. The results of the tests satisfy the assumed limiting energy value, 40 ft·lb (5.5 kg·m), after irradiation to $1 \times 10^{19} n/cm^2$ ($E > 1 \text{ MeV}$).

¹ ΔTr_{30} is the change in NDT temperature as defined at a Charpy V-notch value of 30 ft·lb.

3. A SAFETY STUDY ON REACTOR FUELS

3.1. Behaviour of fuel cladding tubes [21]

3.1.1. Strength and ductility of Zircaloy fuel claddings under loss-of-coolant accident conditions [22]

Out-pile experiments on the behaviour of fuel claddings under loss-of-coolant accident conditions have been conducted. The influence of oxidation in the post-accident atmosphere on the burst strength of tubes containing inert gas simulating a released fission product was also examined.

No effect of corrosion in Zircaloy-2 that should affect its strength could be found up to a weight gain corresponding to several years of use under normal operating conditions either in burst strength or creep burst behaviour.

3.1.2. Effect of non-uniform wall thickness and surface defects on burst test results of fuel cladding tubes [23]

The study was undertaken to evaluate and determine "an acceptable defect level" of fuel cladding tubes for a light-water reactor. The burst strengths of both Zircaloy-2 and AISI 304 L stainless-steel tubes were evaluated by burst-testing naturally or artificially defective tube samples. Artificial defects were created by using a newly designed apparatus [24].

The results showed that: (1) the burst strength of a Zircaloy-2 tube with a non-uniform wall thickness is determined by minimum thickness of the tube; (2) a deep, natural defect on the wall of a tube generally impairs the burst pressure value more than an artificially created defect of the same extent as determined by a non-destructive test; (3) artificial defects in depths up to 30% of the wall thickness do not cause an appreciable deterioration of burst strength.

3.1.3. Directionality change of hydride in Zircaloy-2 tubing due to temperature and stress cycling, and its effect on circumferential ductility [25]

The possibility of reorientation of a hydride platelet under cyclic stress and temperature imposed on the tubing under reactor operating conditions was examined.

Experiments have been carried out, using special devices developed to investigate the effect of temperature cycling between room temperature to 300°C up to 30 cycles, with stresses up to 20 kg/mm² on the Zircaloy tubes, containing 35 - 300 ppm and 1000 ppm of hydrogen, respectively. The results show that stress reorientation was more clearly demonstrated in the case of repeated heating and cooling than a one-cycle experiment, and the length of the hydride platelet decreased with repeated heating and cooling. The change in circumferential ductility of the tubes due to this cycling was evaluated by flare tests, flattening tests, core cylinder expansion tests and burst tests. With the stress of 20 kg/mm², one cycle and five cycles resulted in some decrease of room-temperature ductility. An increase in the fraction of radially oriented hydride is considered to reduce circumferential ductility, whereas a reduction in the size of hydride

is considered to increase the ductility. The ductility reduction noted with one cycle and five cycles may be the result of the combination of these two effects.

3.2. Formation mechanism and the behaviour of methyl iodide in a light-water power reactor²

3.2.1. Introduction

The purpose of this investigation is to clarify the formation mechanism and the behaviour of methyl iodide (1) in nuclear fuels, (2) in high-temperature, high-pressure water, and (3) in the reactor containment of a light-water-cooled power reactor.

3.2.2. Formation of methyl iodide in fuels

It was found that in UO_2 at high temperature, methane is shown to be formed from carbon existing as one of the impurities in the UO_2 . The formation of methyl iodide in a homogeneous uranyl nitrate solution in acetone amounts to 10 - 30% in contrast to an inhomogeneous system of UO_2 -methane mixture, in which less than 1% of the iodine produces methyl iodide. A detailed study of the isotopic ratio indicated that methyl iodide formed in a ratio such as $\text{CH}_3^{135}\text{I}/\text{CH}_3^{133}\text{I}$ is greater than the ratio of $^{135}\text{I}/^{133}\text{I}$ formed by irradiation. This suggests that ^{135}I has a higher probability of producing methyl iodide than ^{133}I through the direct fission recoil process. Isotopic exchange reactions have also been investigated in and out of a reactor environment.

3.2.3. Behaviour of iodine and methyl iodide in high-temperature, high-pressure water

The ratio of iodine transfer from UO_2 into water through recoil to that by leaching has been obtained as a function of temperature. The formation of methyl iodide in high-temperature water with and without a slight amount of methyl alcohol has been measured at various temperatures ranging from 150 to 290°C for a wide range of iodine concentrations. The partition coefficients of iodine and methyl iodide between gas and liquid phases have also been obtained for the same temperature range. Important to note is the fact that the behaviour varies greatly among different iodine concentrations. Typical values of the coefficient for iodine and methyl iodide are 10^4 and 3×10^2 , respectively, under normal operating conditions.

3.2.4. Iodine behaviour in a reactor containment

Formation of methyl iodide through the reaction of radioiodine with paint, neoprene rubber, methane and a mixture of carbon dioxide and hydrogen was investigated under a simulated condition of reactor containment

² Cooperative work carried out by the members of the Fission Product Release Research Committee of the Nuclear Safety Research Association of Japan.

under accident, between 80 and 125°C at the iodide concentration of 10^{-4} to 10^{-9} g/litre. The amount of methyl iodide formed has been very slight, being 1 - 10 ppm of introduced iodine in all cases. Moreover, the methyl iodide decomposes quickly above 70°C. A similar experiment has been carried out by using a simulated containment one-twentieth actual size.

The reaction of iodine with several kinds of aerosols has been investigated. The efficiency of an iodine filter may be reduced by possible reaction between iodine and certain types of submicron particulates in a containment under accident.

4. A PROPOSAL FOR THE SAFETY EVALUATION STANDARD OF NUCLEAR POWER PLANTS [26, 27]

4.1. Introduction

A probabilistic safety evaluation method on nuclear power plants has been studied for several years by the "Expert Committee on Reactor Safety Evaluation and Reliability of Engineered Safeguards" in the Nuclear Safety Research Association.

The method proposed here is to evaluate the risks of a plant for individuals and the public, in general, respectively, for both normal operation and accidents, and to compare such risks to corresponding "Reference Risks".

A concept of individual risk evaluation is shown in Fig. 9 and public risk evaluation should be estimated in the same manner. The basic idea in obtaining "Reference Risks" is as follows.

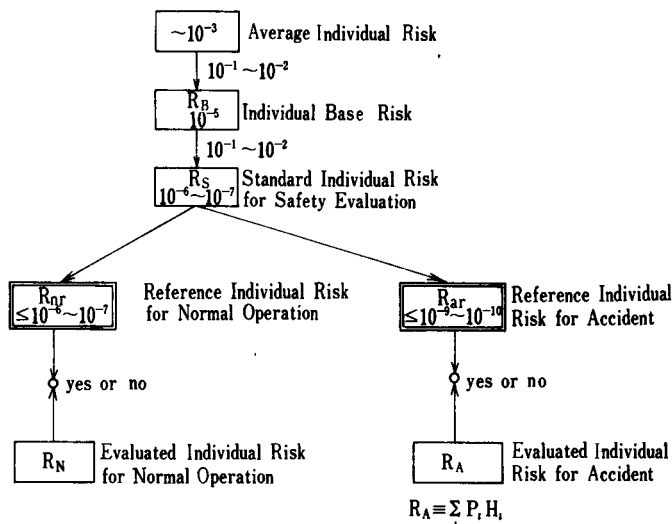


FIG. 9. Concept of Individual Risk Evaluation.

4.2. Individual risks

Human risk data in Japan are shown in Fig. 10. This figure represents public risks, defined as an average of individual risks. The "Individual Base Risk", R_B , is defined as the minimum value of individual risks, the unavoidable risk for each individual.

It is presumed that the "Individual Base Risk" (minimum individual risk) is lower than the public risk (average individual risk) by a factor of 10 to 10^2 . In Fig. 10 the "Individual Base Risk" is assumed as:

$$R_B = 10^{-5} \text{ death/man-year} \quad (1)$$

In establishing the "Standard Individual Risk for Safety Evaluation", R_S , the following items are considered:

(1) Future lessening of the individual risk which decreases R_B .

(2) No significant risk increase by the nuclear plant.

A factor of 10^{-1} to 10^{-2} is assumed for the above considerations and then

$$R_S = 10^{-6} \text{ to } 10^{-7} \text{ death/man-year} \quad (2)$$

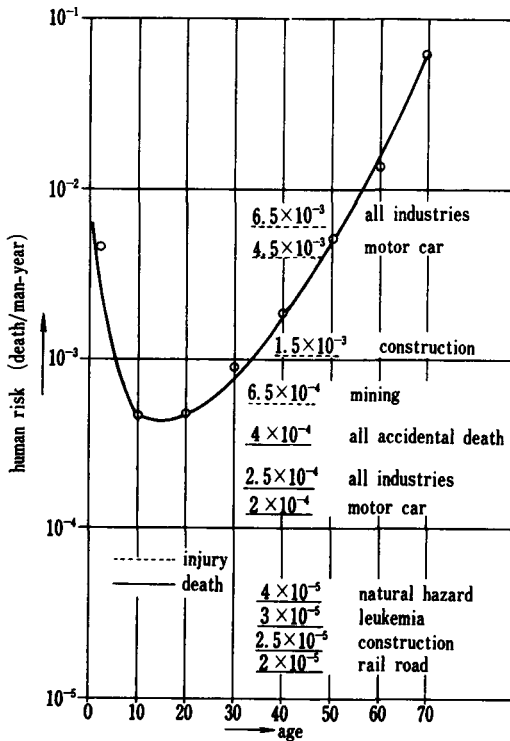


FIG. 10. Human risk data in Japan.

The "Reference Individual Risk for Normal Operation", R_{nr} , and the "Reference Individual Risk for Accident", R_{ar} , are proposed as follows:

$$R_{nr} \leq A_{nr} \cdot R_S \quad (3)$$

$$R_{ar} \leq A_{ar} \cdot R_S \quad (4)$$

where A_{nr} and A_{ar} are application coefficients to the risk in normal operation and in an accident situation, respectively. In normal operation, radioactive waste release is strictly regulated, fully controlled and monitored; thus there remain few uncertainties in probabilistic risk evaluation, and it is reasonable to assume A_{nr} nearly unity, so that:

$$R_{nr} \leq 10^{-6} \text{ to } 10^{-7} \text{ death/man-year} \quad (5)$$

On the other hand, in an accident condition, probabilistic risk evaluation contains many uncertainties and inaccuracies, such as:

- (1) In analysing accidents, some accidents might be overlooked and there is little information on the reliability of a primary system.
- (2) In making reliability calculations, reliability calculation models are inaccurate, failure data of equipment in accident conditions are lacking, and it is very difficult to estimate the reliability of human actions.

In addition, efforts must be made to avoid extremely large accidents even though the evaluated risk (a product of probability and magnitude of hazard) is tolerated.

Taking into account the above-mentioned considerations, a factor of 10^{-3} may be considered for A_{ar} , and the following is then obtained:

$$R_{ar} \leq 10^{-9} \text{ to } 10^{-10} \text{ death/man-year} \quad (6)$$

4.3. Public risks

The same kind of risk evaluation should be performed for public risks. Whereas individual risk evaluation is mainly related to individuals living in the surroundings of a plant, public risk evaluation is concerned with the overall population. So the "Reference Risk" for the public should be established independently of the "Reference Individual Risk", for both normal operation and accidents.

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THE ASEISMIC DESIGN OF NUCLEAR POWER PLANTS IN JAPAN

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Abstract—Résumé—Аннотация—Resumen

THE ASEISMIC DESIGN OF NUCLEAR POWER PLANTS IN JAPAN.

Five nuclear power plants are currently in operation in Japan and ten more are under construction, although Japan is in an area with the highest seismicity in the world. In reactor design and construction, careful consideration is therefore given to aseismic precaution with special attention being paid to the details of various components. This paper presents a method of determining the design ground movements applied to actual power plants, a summary of the criteria for aseismic design now in use, and a proposal for improving present practice. Important subjects for further study are the dynamic analysis of equipment and piping and their allowable stresses under earthquake loading.

LA CONCEPTION ASEISMIQUE DES CENTRALES NUCLEAIRES AU JAPON.

Au Japon, cinq centrales nucléaires sont actuellement en service et dix sont en cours de construction, bien que le Japon se trouve dans l'une des régions à sismicité la plus élevée du monde. Un soin tout spécial est apporté à la conception et la construction des centrales nucléaires, et en particulier à leurs composants, pour les protéger contre les effets des séismes. Le mémoire présente une méthode de détection des secousses sismiques et ses applications dans les centrales nucléaires existantes, ainsi qu'un résumé des critères sur lesquels sont fondées les études parasismiques; il propose également des moyens qui pourraient permettre d'améliorer la pratique actuelle. Parmi les problèmes qui devraient faire l'objet d'études plus approfondies on peut citer notamment l'analyse dynamique des équipements et de la tuyauterie et les contraintes admissibles en cas de séisme.

ДОСТИЖЕНИЯ В ОБЛАСТИ СЕЙСМОСТОЙКОГО ПРОЕКТИРОВАНИЯ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ В ЯПОНИИ.

В настоящее время в Японии введено в строй пять атомных электростанций и сооружается еще десять. Их строительство будет продолжаться и в будущем, несмотря на то что Япония находится в районе земного шара, который наиболее часто подвергается землетрясениям. Поэтому при проектировании и строительстве атомных электростанций принимаются меры, обеспечивающие сейсмостойкость сооружений, при этом особое внимание уделяется элементам различных узлов конструкции. В докладе представлен метод определения сейсмических колебаний, используемый при проектировании, которые учитываются при сооружении атомной электростанции, обобщены критерии сейсмостойкого проектирования, применяемые в настоящее время на практике, и внесено предложение по усовершенствованию современных методов проектирования, что потребует дальнейшего исследования некоторых вопросов. К числу наиболее важных областей исследования относятся динамометрический анализ оборудования и трубопроводов, определение их допустимого напряжения при сейсмической нагрузке.

EL DISEÑO ASISMICO DE CENTRALES NUCLEOELECTRICAS EN EL JAPON.

En el Japón existen actualmente cinco centrales nucleoelectricas en funcionamiento y se hallan en construcción otras diez, a pesar de que ese país se encuentra situado en una zona que presenta la mayor intensidad sísmica del mundo. Por lo tanto, el diseño y la construcción de las centrales han recibido una atención muy particular, especialmente en lo que respecta a sus componentes, a fin de protegerlas contra los efectos

de los sismos. La memoria describe un método para determinar los factores parasísmicos que deben tenerse en cuenta al establecer el diseño de las centrales y contiene un resumen de los criterios de diseño sísmico actualmente en uso y propuestas para mejorar las prácticas existentes. Entre los problemas que deberfan estudiarse más a fondo figuran los análisis dinámicos de equipo y tuberías, y su resistencia límite bajo cargas sísmicas.

1. INTRODUCTION

The first commercial nuclear power plant in Japan was the Tokai Power Station (166 MW(e)) of the advanced Calder Hall type, which was completed in 1965. At present, three light-water type reactor (BWR and PWR) stations are in operation and ten power plants are under construction.

Design practices for nuclear power plants have been changing and evolving on the basis of continuous research. In an attempt to establish provisional aseismic design criteria, JEA (Japan Electrical Association) proposed "Technical Guidelines for Aseismic Design of Nuclear Plants" in October 1970, after three years of intensive work. The ACRS (Advisory Committee on Reactor Safety) of Japan adopts a more conservative attitude in granting licences than does the JEA group, and does not entirely accept the above Guidelines. The major difference between the two groups is the way in which the definition of the maximum design earthquake is interpreted. In this paper, Japanese design practices are outlined and the different interpretations are discussed.

2. FUNDAMENTALS

2.1. Object of aseismic design

The object of aseismic design for nuclear power plants is to ensure the absolute protection of both the public and the plant personnel from radioactivity released during and after a severe seismic disturbance.

As stated in the IAEA report of 1968 [1], the basic design philosophy in seismic countries is:

- (1) nuclear power plants built in seismic zones should be designed and constructed in such a way that they produce no significantly greater radiation hazard than nuclear power plants built in non-seismic zones;
- (2) the degree of conservatism in providing for the possible effects of earthquakes should deliberately be taken as higher than that used in the design of other types of facilities and plants.

All structures, equipment, piping and instrumentation are carefully designed along these lines to avoid earthquake damage; for structures and elements whose failure might cause radiation hazard and for systems to prevent this hazard from spreading, strict regulations are imposed in the earthquake resistant design.

2.2. Classes of aseismic design

To attain the above object, all buildings, structures, equipment, piping and instrumentation are classified into the following four classes according to their degree of importance for plant safety:

- A_s class: Essential items such as reactor containment and safety shutdown devices.
- A class: Items whose functional failure would lead to an occurrence of nuclear accidents, and items required to protect the public from nuclear hazards in the event of a nuclear accident.
- B class: Items related to highly radioactive materials, except those in the A_s and A classes.
- C class: General items not included in the A_s, A and B classes.

3. DESIGN EARTHQUAKES

3.1. Definitions

As mentioned in the Introduction, there are differences of opinion over the definition of the design earthquake between JEA and the ACRS.

3.1.1. JEA definition

(1) With respect to intensity for the design of A_s and A class items, the maximum design earthquake is defined as follows:

- (a) the intensity should be greater than that which can be estimated from data of past earthquakes and seismic disasters in the area around the specific site;
- (b) the intensity should be the upper limit of the maximum intensity which would be expected to occur in the area within the next several hundred years.

It can be supposed that the probability of occurrence of the above defined maximum design earthquake in the area is very low, but this earthquake should be assumed to occur with a certain probability during the plant lifetime.

(2) In addition to the above maximum design earthquake, the other hypothetical earthquake to establish the adequacy of the safety functions of A_s class items and of the containment and safety shutdown systems is defined as 150% of the maximum design earthquake intensity.

3.1.2. ACRS definition

Because of the uncertainties involved, two earthquake intensities are assumed to act at the base of the reactor building for the purpose of dynamic analyses. These two design earthquakes of different intensities are referred to here as Class A design earthquake and Class A_s design earthquake.

(1) Intensities of Class A design earthquakes that have actually been used so far range from 180 to 300 gal, according to seismic situations in different sites, and the value of 180 gal is accepted as the lower limit of design intensities in Japan. The provision of such a lower limit is considered desirable until the seismo-tectonic procedures for intensity evaluation of future earthquakes are more reliably and precisely established. In establishing the lower limit, design practices in this country for other types of important structures such as long-span bridges, high buildings, large dams, etc., were deliberately considered to ensure that the safety criteria for nuclear facilities would be more conservative.

(2) The Class A_s design earthquake is chosen to ensure an adequate margin of safety for Class A_s items, i.e. the most vital components such as containment structures, safety shutdown devices, etc. In current practice, the intensity of this design earthquake is usually taken as 1.5 times that of the Class A design earthquake. It should be emphasized that, from a conservative point of view, even this Class A_s design earthquake has a certain probability of eventual occurrence; at present, no-one can make a definitive prediction as to the intensity of the strongest earthquake.

3.2. Evaluation of seismic expectancy

Under the assumption that earthquakes will take place as a statistical phenomenon, it is possible to calculate the probability with which each location is attacked by an earthquake having some definite intensity.

Kawasumi [2] proposed equations by which an intensity can be calculated at the distance Δ when the magnitude of an earthquake is known. They are:

$$I = M_k + 4.605 \cdot \log_{10} (100/\Delta) - 0.00183 (\Delta - 100) \quad \Delta > 100 \text{ km} \quad (1)$$

$$= M_k + 2 \log_{10} (R_0/R) + 2k \cdot \log_{10} e \cdot (R - R_0) \quad \Delta < 100 \text{ km} \quad (2)$$

where M_k : Kawasumi Magnitude (JMA intensity grade at the epicentral distance of $\Delta = 100$ km)

R : Hypocentral distance

R_0 : Hypocentral distance where $\Delta = 100$ km

k : Attenuation constant for shear waves; usually is given by 0.0192 (per 1 km).

For all large earthquakes in the past, Kawasumi determined the expected intensity at each location by means of the above equations.

To relate the intensity I to the acceleration, Kawasumi used the following equation:

$$\alpha = 0.45 \times 10^{I/2} \text{ gal} \quad (3)$$

From this equation it is easy to calculate an acceleration at each location when I is given in Eq. (3). It should be noted here that Eq. (3) is based on the data of seismograph records of the Earthquake Research Institute, University of Tokyo.

Similarly, the expected maximum velocity amplitude in bedrock at each location has been worked out by Kanai [3]. In this case, the velocity V_0 (cm/s) in bedrock was calculated with the following equation:

$$\log_{10} V_0 = 0.61 M - \left(1.66 + \frac{3.6}{R}\right) \log_{10} R - \left(0.631 + \frac{1.83}{R}\right) \quad (4)$$

where R represents the hypocentral distance in kilometres.

3.3. Uncertainties associated with earthquakes

Considerable importance is attached at present to an approach on a statistical or probabilistic basis in an attempt to attain a seismic evaluation for a given area. It should be recalled, however, that in recent years (1964 and 1968) two destructive earthquakes took place in regions of the lowest seismic probability, indicating that Japanese seismic history, which has extended over one thousand years, is not long enough to define a reasonable and reliable intensity of a maximum earthquake for nuclear power-plant design.

Fortunately, a recent trend is towards applying a seismo-tectonic approach to the estimation of potential locations and magnitudes of large earthquakes which, in future, might influence the safety of a specific power plant. Because of the complexity of the geological and tectonic structures of the Japanese islands and a lack of precise knowledge in this science, this approach is still far from being established and is not necessarily accepted generally, even among seismologists in Japan. Nevertheless, it is highly promising and should be encouraged, since it will contribute to a deterministic evaluation of design earthquakes as a satisfactory substitute for an uncertain, probabilistic estimate.

3.4. Estimation of the design earthquake

An earthquake is a phenomenon involving many unknowns, including the location of the epicentre and the intensity of the earthquake motions. Not only from the scientific point of view, but also from the engineering side, there are many problems to be solved in the future. The design earthquake motion to be adopted for the design of nuclear power facilities should therefore be decided on the basis of all the factors mentioned above, and should be arrived at with the aid of research and available experience on earthquake phenomena. The ways in which the evaluation of the design earthquakes is done at present are given below.

- (1) Investigation of seismicity in the area where the site is located.
- (2) Investigation of the history of earthquake damage experienced in and around the site.
- (3) Earthquake observation in the field for the purpose of obtaining the characteristics of the earthquake motion of the ground at the site.
- (4) On the basis of data obtained from investigations on items (1) and (2), the maximum velocity or acceleration of the bedrock at the site is estimated. When abundant data are available, the maximum velocity and acceleration are estimated with respect

to the earthquake that has had the greatest effect on the site in the past, and, where little historical information is available, the magnitude and epicentre of a future earthquake is estimated that would have the greatest effect on the site, with all seismic conditions around the site being taken into account.

- (5) For input earthquake waves, the accelerograms of large earthquakes recorded on hard rock or simulated random impulses may be the most appropriate for calculating the dynamic response of nuclear facilities. In using such earthquake waves, the time abscissa should be adjusted so as to represent the ground characteristics of the site, and the maximum acceleration should also be adjusted to the value determined from item (4).

4. ASEISMIC DESIGN FOR BUILDINGS AND STRUCTURES

4.1. Design methods

4.1.1. Dynamic analyses

For the aseismic design of buildings and structures belonging to Classes A and A_s, dynamic analyses are performed. As described previously, Class A and Class A_s design earthquakes are considered in these dynamic analyses.

In the method of analysis, procedures based on time-history accelerograms and on response spectrum concepts are currently being used to compute induced stresses in major structural elements.

For the computational model of buildings and structures at a nuclear power plant, a discrete lumped-mass system is usually employed in dynamic response analyses, developed so as to simulate the behaviour of the prototype as exactly as possible in an engineering sense and, at the same time, to be as simplified as possible in view of the limited computer capacity. Usually, the effect of the foundation rock in respect to the dynamic response of structures is incorporated in the lumped-mass system.

4.1.2. Static analyses

In addition to the dynamic analyses in which the design earthquakes mentioned above are used, static analyses are also performed as a design practice, based on the conventional concept of the seismic coefficient. The horizontal seismic coefficient is equal to βC_0 , where C_0 is specified basically as 0.2 and can be reduced to 80% of the specified value when the base of the structure is on firm rock. β is 3.0, 1.5, and 1.0 for Class A, B, and C items, respectively.

4.1.3. Vertical seismic force

For all cases, the simultaneous action of vertical seismic forces is considered and assumed to be one half of the horizontal forces. Furthermore, the vertical forces should be applied in the upward or the downward direction, whichever is more disadvantageous, when combined with the horizontal direction.

4.1.4. Determination of working stresses

Dynamically and statically determined stresses at critical points are calculated and compared, and the major structural components are designed to withstand whichever are the greater. The additional application of a static analysis is intended to compensate for uncertainties, if any, in evaluating the dynamic characteristics of the structural elements and for inadequate modelling of the structures in the dynamic analyses.

4.2. Loading combinations and allowable stresses

Seismic load is combined with the dead load, the live load and, if necessary, the snow load in deep snow areas in the aseismic design for buildings and structures other than the containment. Allowable stresses for these buildings and structures are primarily based on the Japanese Building Code [4] and the AIJ Structural Standards [5]. In these Codes and Standards, it is required that stresses in steel remain within the elastic range, and particular attention is paid to possibilities of shear failure of concrete.

5. ASEISMIC DESIGN OF EQUIPMENT AND PIPING

5.1. Design conditions

The plant safety depends largely on the functioning of equipment and pipings, which must be carefully designed to withstand earthquake motions. Class A items are designed by a method based on dynamic response analyses. Several computer codes were developed and new criteria for load combinations and allowable stress were proposed.

5.1.1. Class A items

In designing equipment and pipings, the following earthquake forces are considered:

- (1) The design seismic force is determined from the dynamic analysis that takes into account the vibration characteristics of equipment, pipings and buildings, and in which either time-history accelerograms or response spectrum curves for the maximum design earthquake are used.
- (2) The force calculated is not to be less than the value obtained by increasing (to not less than 120%) the design horizontal seismic coefficient at the elevation of the supporting point in the building which is assumed to be Class A.
- (3) The vertical earthquake force used for Class A structures is also considered to act simultaneously. Class A_s items are examined to ensure their safety during an earthquake which is not less than 150% of the maximum design earthquake.

5.1.2. Class B items

The design horizontal earthquake force for each item is determined by the seismic coefficient which is taken to be 1.2×1.5 times of the basic value, C_0 , at the elevation of the supporting point in the building which is assumed to be Class B. Equipment and pipings are designed on the basis of dynamic analysis according to their vibration characteristics.

5.1.3. Class C items

For Class C, aseismic design is carried out when necessary.

5.1.4. Floor response curves and earthquake waves

To draw a floor response curve chart or to evaluate the time-history response of equipments and pipings, three or four time-history accelerograms are used: two of the famous records, for example El Centro NS, 1940, and Taft S69 E, 1952, one of the records obtained on rock foundation, for example Golden Gate, 1957, and the records obtained at the site. If the base rock of the site is rather soft, the Golden Gate type can be omitted. Records in which the maximum accelerations are less than 25 gal at the site might not represent future strong earthquake motion.

The standard floor response curve for the design may be drawn by making an envelope of the response curves of the specified earthquakes.

5.2. Practical design methods

5.2.1. Computing technique

Several computer codes were developed to evaluate the vibration characteristics, the responses and the stress distribution of equipment and piping.

Some of them are based on the results of the study of the sub-committee on "Aseismic Design of Nuclear Facilities" in the Japan Society of Mechanical Engineers from 1962 to 1969, including the development of the codes DYNAPS and SHEL VIA. DYNAPS is a code for the eigenvalue and mode shape analysis of a piping system and incorporates a transfer matrix method.

To obtain the distribution of maximum stress values of piping systems under a strong earthquake, the modal analysis technique is usually used with the design floor response curve. The direct time-history response technique is rather seldom used. Critical damping ratios of 2% and 1% are generally employed for the response analysis of piping systems and vessels, respectively.

5.2.2. Special devices

Typical earthquake resistant devices used in boiling-water type reactor plants are described below.

(1) Shear-lug resistance: The bottom of the drywell or the containment vessel, which is installed within a reactor building, is fixed to the foundation concrete. The top is laterally jointed to the surrounding concrete structure. To allow the vertical and radial movement of the drywell caused by thermal expansion and also for other reasons, several radial keys are provided at the top ring of the drywell. The bottom of the torus is also fixed to the reactor building through radial keys.

(2) Pre-tension mechanism: The top of the pressure vessel is laterally supported by the ring of the internal concrete structure through a pre-tension mechanism. The ring is connected through another pre-tension mechanism to the drywell at the positions of the radial keys. The pre-tension mechanism consists of tension rods, which are mounted in circumferential direction, and of turn buckles.

(3) Hydraulic damper: Hydraulic dampers or snubbers, lock mechanisms for a sudden movement by an earthquake, are used as part of the supporting mechanisms of piping systems. The idea of using hydraulic snubbers originated with the construction of the Tokai Power Station. The large gas ducts are supported by counterweight type constant hangers, and each counterweight is tied back to the duct by hydraulic snubbers.

5.3. Loading combination: current practice and a new proposal for allowable stresses

5.3.1. General

Seismic load is combined with the dead load, live load and thermal load in the aseismic design for equipment and piping. ACRS requires the allowable stresses for such items made of steel to remain within the elastic range or not significantly to exceed the yield point. ACRS also requires that the ductility be considered so that conditions approaching failure or collapse can be avoided, if yielding is to be permitted in any section or component of elements.

5.3.2. Containment vessel

For the containment vessels, the following two loading combinations should be considered to assure integrity:

- (a) normal operating load, accident load, Class A design earthquake load;
- (b) normal operating load and Class A_s design earthquake load.

Under these loading conditions, the primary stresses are not permitted to exceed significantly the yield point of the material so that excessive deformation does not take place and the containment can maintain its function as the last barrier between the radioactivity and the external atmosphere. The necessity of loading condition (a) as one of the design criteria is based on the possibility of occurrence of an extreme earthquake for the duration of an accident.

5.3.3. New proposal

A new proposal [6] was issued in the Guidelines very recently by JEA. In this proposal, the allowable stress values are determined so as to meet the MITI code and the aseismic design criteria that the plants must at least remain in emergency conditions under the combination of loadings in normal operation and in the maximum design earthquake, i.e. under the specified earthquake conditions there must be certainty that no gross loss of structural integrity will result as a concomitant effect of any damage in the system, so that the system shall retain its function and its operational status shall be regained with repair of the damage.

The proposal for allowable stress values is given below.

(1) Class 1 vessel in MITI code

The stress intensity limit for the primary general members stress P_m is the greater of $1.2 S_m$ and S_y , where S_y is yield strength and S_m is the smaller of $2/3 \cdot S_y$ and $1/3 \cdot S_u$, S_u being the ultimate tensile strength, and the limit for the primary local component stress plus the primary bending stress $P_L + P_b$ is the greater of $1.8 S_m$ and $1.5 S_y$ or $0.8 C_L$, where C_L denotes the collapse load calculated by limit analysis based on yield strength S_y . As for the secondary stress, the cyclic range of the sum of the primary and the secondary stress $(P_L + P_b + Q)_E$ due to the earthquake loading alone is limited to less than $3 S_m$. When the secondary stress value exceeds $3 S_m$, fatigue analysis is required. The sum of the principal stresses in tension $\sigma_1 + \sigma_2 + \sigma_3$ is limited to less than $4.8 S_m$.

(2) Class 2 vessel in MITI code

Detailed stress analysis shall be made only in the region of high stress due to earthquake loading, and the primary stress P_m is limited to less than the greater of $1.2 S$ and $15/16 \cdot S_y$, and $P_L + P_b$ is limited to less than the greater of $1.8 S$ and $1.5 \times 15/16 \cdot S_y$ or $0.8 C_L$, where S is the smaller of $5/8 \cdot S_y$ and $1/4 \cdot S_u$. The secondary stress range $(P_L + P_b + Q)_E$ due to earthquake loading is limited to less than $3 S_m$, and when it exceeds $3 S_m$, fatigue analysis is required.

(3) Class 3 and 4 vessels in MITI code

The maximum stress, generally primary stress, is limited to less than the greater of $1.6 S$ and S_y .

(4) Class 1 piping

The primary stress value calculated by the formula

$$B_1 \frac{D_o P}{2t} + B_2 \frac{D_o M_1}{2I}$$

is limited to less than $1.8 S_m$, where P = design pressure, D_o = outside diameter of pipe, t = wall thickness, I = moment of inertia, M_1 = resultant moment due to weight, mechanical load and earthquake loading, and B_1 , B_2 = primary stress indices. At the same time, the stress value calculated by

$$B'_2 \frac{D_o T}{2I}$$

is limited to less than $1.4 S_m$, where T = twisting component of M_1 , and B'_2 = primary torsional stress index. However, the latter stress limitation is unnecessary when the axial stress due to internal pressure exceeds $2/5 S_m$. The cyclic range of the secondary stress calculated by

$$C_2 \frac{D_o}{2I} (M_1)_E$$

due to the earthquake loading alone is limited to less than $3 S_m$, where $(M_1)_E$ = range of resultant moment due to earthquake loading alone and C_2 = secondary stress index; when the secondary stress value exceeds $3 S_m$, fatigue analysis is required.

(5) Class 2 piping

The maximum stress is limited to less than the greater of $1.6 S$ and S_y .

(6) Support and structural components

- (a) Functional component: the allowable stress value is the same as that relating to equipment and piping.
- (b) Structural component: for a component for which a stress analysis is made as for a Class 1 vessel, the allowable stress limit is the same as for a Class 1 vessel. For other structural components, the allowable stress value specified in the AIJ Structure Standard [5] is applied.

(7) Bolts

The allowable stress value is the same as for support and structural components.

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EARTHQUAKE PROTECTION FOR NUCLEAR REACTORS

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Abstract-Résumé-Аннотация-Resumen

EARTHQUAKE PROTECTION FOR NUCLEAR REACTORS.

It is not always possible to site nuclear plants in earthquake free zones. Designs have therefore to be developed which can withstand earthquakes of a given magnitude so that the safety of the plant will not be jeopardized and that damage will not result which could cause the plant to be out of service for appreciable periods. It is possible to achieve the desired result either by designing every component to withstand the forces occurring during the given earthquake, or to devise a means of reducing the resultant forces by building the plant on some form of damping medium or mechanism.

It is usual in earthquake zones to incorporate features in the design to accommodate the resultant forces and this solution is also generally adopted in the case of nuclear power plants. Designing for safety and reliability in earthquake zones in this manner can, however, add considerably to the overall cost of the plant and there is thus an incentive to study other methods. Mounting structures on rubber or other damping media has been proposed and used, and at Brown Boveri we have developed a mechanical system to reduce the resultant forces. The paper outlines the Brown Boveri development, which, we believe, will allow nuclear power plants or any other type of structure to be sited safely and economically in earthquake zones.

PROTECTION DES REACTEURS NUCLEAIRES CONTRE LES SEISMES.

Il n'est pas toujours possible d'implanter les centrales nucléaires dans des zones à l'abri de tout tremblement de terre. On a dû ainsi étudier des méthodes de construction telles que les centrales puissent résister à des tremblements d'une certaine ampleur sans que leur sécurité en soit affectée et sans que leurs systèmes courent le danger d'être mis hors service pour de longues périodes. On y parvient soit en dimensionnant chaque composante de manière qu'elle résiste aux forces du séisme, soit en trouvant le moyen de réduire les forces résultantes en bâtissant la centrale sur une structure ou un mécanisme amortisseur.

Cette dernière méthode est habituelle dans les zones à séismes. Elle est généralement adoptée pour les centrales nucléaires. Il en résulte cependant des frais supplémentaires, d'où l'intérêt de trouver d'autres méthodes. Le montage sur des blocs de caoutchouc ou d'autres matériaux amortisseurs a été proposé et utilisé. Brown Boveri a développé un système mécanique pour réduire les forces résultantes. Le mémoire présente des travaux de recherche de Brown Boveri dans ce domaine, qui devraient permettre l'installation sûre et économique d'usines atomiques ou d'autres constructions dans les zones exposées aux séismes.

ЗАЩИТА ЯДЕРНЫХ РЕАКТОРОВ ОТ ЗЕМЛЕТРЯСЕНИЙ.

Иногда приходится строить ядерные установки в районах, подверженных землетрясениям. Поэтому необходимо разработать такие конструкции, которые могли бы выдерживать землетрясения определенной силы, не допуская снижения степени безопасности установки и возможности возникновения таких повреждений, которые могли бы привести к выходу установки из строя на длительные периоды времени. Этого можно достичь либо путем разработки всех узлов установки с таким расчетом, чтобы они могли выдерживать действие сил, возникающих во время конкретного землетрясения, либо путем изобретения какого-то средства, которое позволило бы уменьшить действие возникающих при землетрясении сил, построив, например, установку на амортизирующей среде или системе.

Обычно разработка проектов для районов, подверженных землетрясениям, осуществляется с учетом тех сил, которые возникают во время землетрясения, и этот принцип общепринят также при конструировании атомных электростанций. Проектирование с учетом

обеспечения безопасности и надежности работы установок в районах, подверженных землетрясениям, может, однако, значительно повысить общую стоимость установки, и это стимулирует поиски других методов достижения желаемой цели. Была предложена и использована крепежная конструкция, установленная на резиновом или другом амортизирующем основании, а фирмой "Браун-Бовери" была разработана механическая система, снижающая воздействие сил, возникающих при землетрясении. В докладе дается описание системы, изготовленной фирмой "Браун Бовери", которая, как полагают авторы, обеспечит безопасное и экономичное строительство атомных электростанций или других сооружений в районах, подверженных землетрясениям.

PROTECCION DE LOS REACTORES NUCLEARES CONTRA LOS TERREMOTOS.

No es siempre posible situar las centrales nucleares en zonas sísmicas. Por ese motivo hay que desarrollar diseños capaces de resistir los terremotos de una determinada magnitud de modo que no resulte afectada la seguridad de la central y que no se produzcan daños que puedan dejar la central fuera de servicio durante largos períodos. Puede conseguirse el resultado deseado proyectando cada componente para resistir las fuerzas originadas durante un determinado terremoto, o bien ideando un procedimiento que reduzca las fuerzas resultantes mediante un medio o mecanismo amortiguador sobre el que vaya construida la central.

En las zonas sísmicas, es frecuente que los proyectos se adapten a las fuerzas resultantes y esta solución se ha adoptado también, en general, en el caso de las centrales nucleares. Sin embargo, realizar los proyectos de este modo en las zonas sísmicas, con los criterios de seguridad y fiabilidad, puede suponer un aumento considerable en el costo total de la central, por lo que hay buenas razones para investigar otros métodos. Se ha propuesto (e incluso se ha utilizado), instalar la estructura sobre caucho u otros medios amortiguadores y en la compañía Brown Boveri se ha desarrollado un sistema mecánico que reduce las fuerzas resultantes. En la memoria se bosqueja el desarrollo de este procedimiento de Brown Boveri que, en nuestra opinión, permitirá instalar centrales nucleares o cualquier tipo de estructura en zonas sísmicas y en buenas condiciones de seguridad y economía.

1. INTRODUCTION

Consideration of the behaviour of a graphite-moderated gas-cooled high-temperature reactor (HTR) during earthquakes forms a major part of the HTR work being carried out by Brown Boveri. Since safety measures against earthquakes had already been established in connection with the design of other nuclear power plants it was logical to apply this experience also to the HTR.

After a short summary of the existing techniques, this paper shows why these methods could not be applied to the HTR and how the problem was solved despite this difficulty.

The following study uses the 'reference earthquake' established by experts for the nuclear power plant site which was taken as the reference site for the HTR studies. The 'reference earthquake' is defined as follows:

Max. earthquake acceleration:

(g = acceleration due to gravity)

horizontally 0.12 g

vertically 0.08 g

Max. amplitude

within the frequency

range of: 3-10 Hz

2. PRESENT STATE OF KNOWLEDGE ON EARTHQUAKE PROTECTION

Most available techniques rely on stiffening of the components or buildings to be protected. Static or dynamic analysis is used to establish the degree of strengthening required.

2.1. Static method

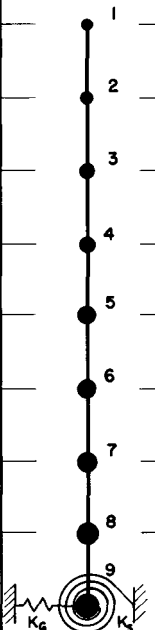
The simplest case assumes that the acceleration due to the earthquake is constant over the entire height of the building. This assumption is considered to be unrealistic for nuclear power plants.

In a more realistic case, it is assumed that the acceleration increases linearly with the height of the building; for example, at 40 m above ground the acceleration is twice that at ground level. Extensive studies based on exact dynamic analysis have confirmed this assumption in certain cases.

2.2. Dynamic method

2.2.1. First approach

The first approach was to study the behaviour of the systems when the complicated earthquake spectrum was replaced by simple harmonic oscillations. This did not work, however, because the natural frequencies

Building		Mathematical Model	Section Properties	
Elevation (m)	Weight (Tons)		I_c (m ⁴)	A_c (m ²)
59.5	309		4 900	26
53.5	964		6 673	36
49.7	812		6 673	36
41.9	5338		16 336	212
34.0	5 738		16 252	201
28.5	5 980		14 868	146
20.5	7 386		13 619	183
12.5	9 484		24 435	235
0	27 171			

Base: $A = 1\,598\text{ m}^2$

$I = 203\,341\text{ m}^4$

Ground: $K_s = 44\,275\text{ T/m}^3$

$K_g = 32\,501\,331\text{ T/m}^3$

FIG. 1. Scheme of typical reactor building.

of heavy components are mostly within the critical frequency range of the reference earthquake, leading to unrealistic resonance cases, and thus, theoretically, to failure of the components being studied.

2.2.2. Behaviour of multi-mass systems

In the ideal case the ground, the foundation, the building, and the various components in the building can be represented by a number of mass points connected by springs and damping elements. The response of this model is then studied under the influence of recorded earthquakes normalized to the reference earthquake of the particular site.

A mathematical model for a typical reactor building is shown in Fig. 1. Figures 2 and 3 show the horizontal accelerations and displacements versus the height of the building.

2.2.3. Behaviour of single-mass systems

If the oscillation behaviour of a component can be represented by a single-mass model, the effect of the entire earthquake spectrum on this component can be determined in a simple manner from the 'acceleration response spectrum curve' (Fig. 4). This method was devised by Housner [1].

If the natural frequency (f_1) of the component, its material-dependent damping, and the acceleration at its location (a) are known, the maximum acceleration (a_1) on the component can be determined directly from the appropriate curve. If the determination of the natural frequency is too complicated or doubtful, the maximum value of the curve can be used.

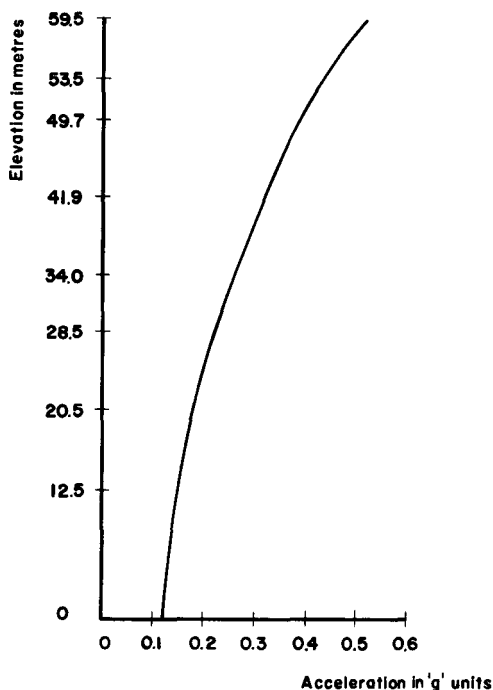


FIG. 2. Maximum acceleration diagram.

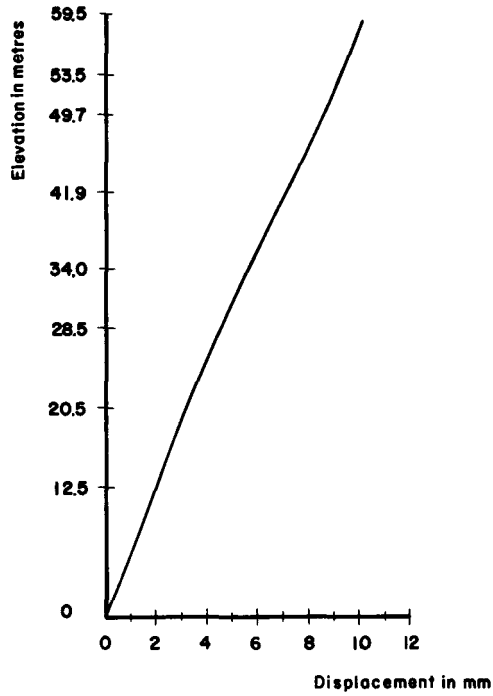


FIG. 3. Maximum displacement diagram.

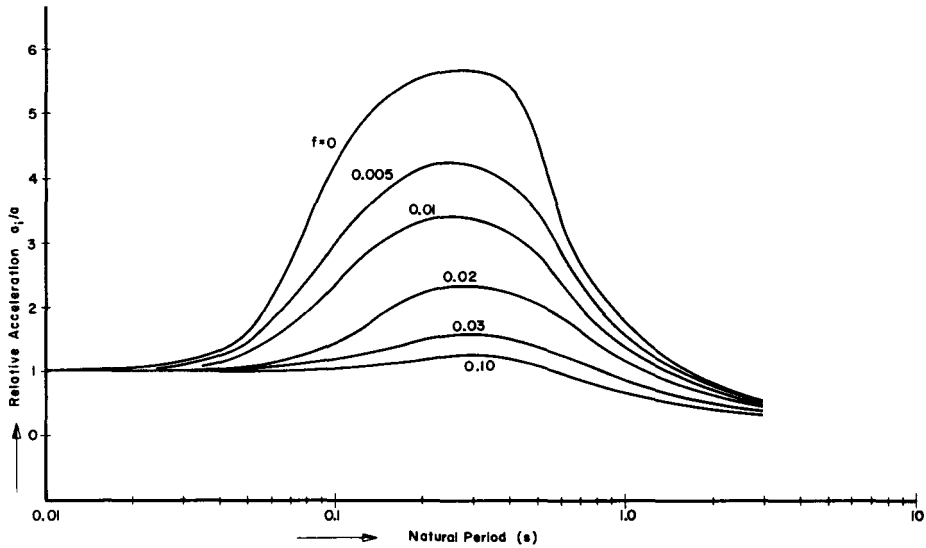


FIG. 4. Acceleration response spectrum curves.

2.3. Limits of application for the described methods

Vertical earthquake components generally present no particular problems, except in a few cases such as crane beams, because buildings are normally strong enough in the vertical direction to carry the acceleration loads and to prevent resonance amplification.

The methods described above, which have been proved to be basically sound, assume first, that the components can oscillate elastically and do not show discontinuities in their motion due to shocks leading to excessive stresses, and second, that it is feasible to protect the components by reinforcing them in such a way that they can withstand the earthquake.

A typical example where these conditions are met is the core arrangement of an HTR in which the individual prismatic moderator blocks are stacked on top of each other, forming free standing columns. Simplified calculations show that during an earthquake shock effects with an intensity of 10 to 20 g might occur due to the necessary clearances between the columns. It is extremely difficult to formulate the law of motion for the type of multi-body shock problem which occurs when the entire core is considered.

Since the principles of building up the HTR core are dictated by a number of considerations, it was not desirable to modify the core itself, so a solution had to be found which would prevent as far as possible the transmission of ground oscillations to the endangered objects.

3. DYNAMIC EARTHQUAKE PROTECTION OF BUILDINGS

3.1. Basic considerations

The stresses on buildings and components during an earthquake depend on the amplitude and frequency of the motion, which are related as follows:

$$A_i \text{ is a function of } A_E, \frac{f_E}{f_i}, \alpha_i$$

where

A_i = amplitude of the component i

A_E = amplitude of earthquake

$\frac{f_E}{f_i}$ = ratio of earthquake frequency f_E to natural frequency f_i of component i

α_i = damping factor of component i .

A_E and $\frac{f_E}{f_i}$ correspond to the time-dependent earthquake spectrum.

To keep A_i small the following possibilities exist.

- (1) Reduce the amplitude of the earthquake A_E by:
 - (a) relocation of site
 - (b) improvement of ground
- (2) Increase the damping factor α_i by:
 - (a) change of material, for example concrete instead of steel
 - (b) addition of damping material or shock absorbers

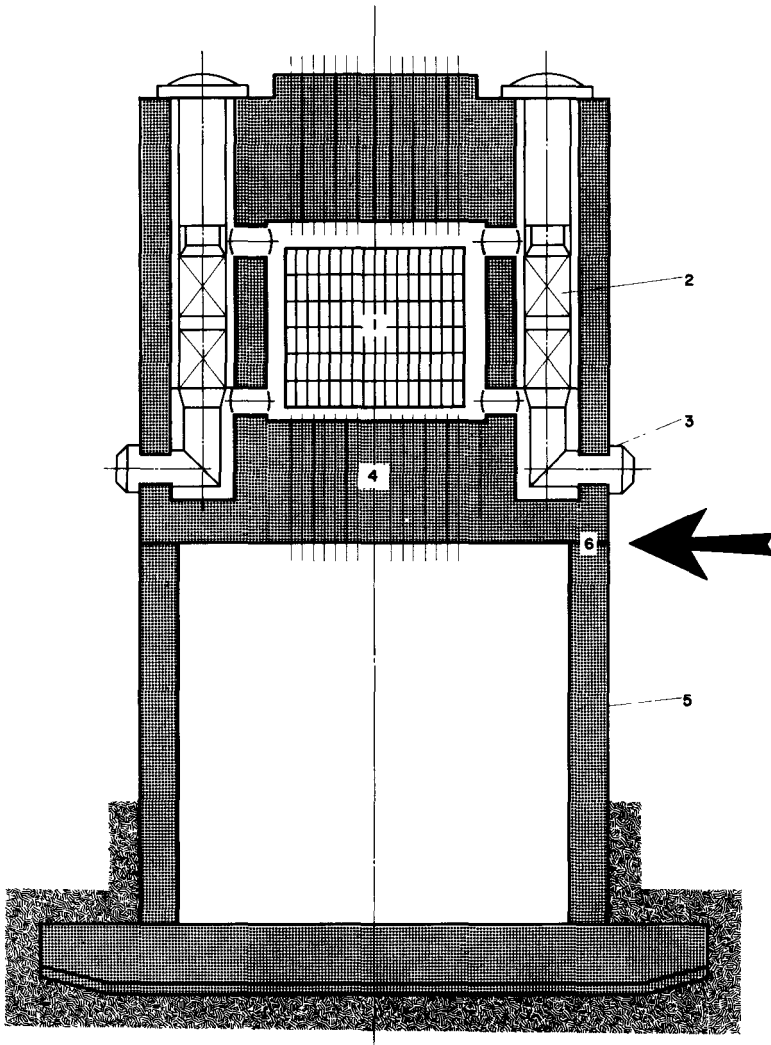


FIG. 5. Schematic section through a nuclear reactor.

- (3) Shift the frequency ratio f_E/f_i away from the resonance range by:
- (a) increasing the natural frequency by stiffening the component:
 $f_i > f_E$
 - (b) reducing the natural frequency of the component to come into the super-critical range of the system: $f_i < f_E$

On analysing the various possibilities it becomes obvious that: the reduction of A_E is a matter of site approval, increased damping is only applicable in a limited manner, and increasing the natural frequency is the objective of the previously mentioned solution.

The remaining possibility, i. e. reducing the natural frequency, is the subject of the following chapters. Protection is achieved by horizontally movable supports which actually carry the building. The oscillation response of this system is displaced far enough into the super-critical range to ensure that resonance between earthquake and building is prevented. The amplitude and accelerations transmitted from the foundation to the building are dramatically reduced. The dynamic response of this system and the influence of damping will be discussed. The extent of protection which can be achieved will be shown, using a HTR project as an example. The mathematical derivations can be found in Ref. [2].

3.2. Effect of earthquakes on the unprotected building

Figure 5 represents the reactor schematically. Its core (1), built of prismatic graphite blocks, the heat exchangers (2), and the gas circulators (3) are located in a pre-stressed concrete pressure vessel (4), which rests on a cylindrical foundation (5). The rocking supports (6), described in subsection 3.3, are located between the upper end of the foundation wall (5) and the concrete pressure vessel (4), indicated by the arrow in the illustration.

The unprotected building can be represented by a single-mass model as shown in Fig. 6(a).¹ The heavy concrete pressure vessel is the oscillating mass (m_p), and the circular foundation wall the spring. The horizontal motion induced by the earthquake at the elastically embedded foundation excites the system.

Figure 6(b) shows the increase of acceleration as a function of the height of the building for this model. The limiting curve on the left of the shaded area represents an ideally rigid embedment of the foundation, while the curve on the right corresponds to very unfavourable and pessimistic assumptions in terms of ground damping and resonance between ground and foundation.

In reality the buildings are always located within the shaded area and, in practice, the two limiting curves would never be reached.

The following two examples show the considerable influence of the ground.

During the earthquake of 28 December 1908 the section of the city of Messina built on second ice-age alluvial soil was the most severely

¹ For purposes of comparison Figs 6(a) and 6(b) are printed above Figs 10(a) and 10(b) on p. 320.

damaged, whilst other sections of the city built on soil of ice-age origin showed less destruction, and buildings erected on crystalline soil showed only slight damage.

During the earthquake of 25 January 1946, damage in Sierre (Switzerland) was limited essentially to those buildings erected on accumulated rubble deposited by a side-arm of the river Rhône, and on a former swamp area.

3.3. Earthquake safety: principle of dynamic protection

Since, in view of the above facts, it is not possible to avoid oscillation of the foundation, the interposition of rocking supports between foundation and building, whose oscillating response can suitably be selected, serves to reduce considerably the transmission of the foundation acceleration to the building.

The principle of the rocking support is shown in Fig. 7. It forms an oscillating system, slightly damped in the horizontal direction, whose basic response can be calculated from the known equation of motion under the following three assumptions:

(a) Damping is proportional to speed and acting parallel to the spring. The constant rolling friction has been approximately converted to an equivalent damping proportional to speed by assuming equal braking energy.

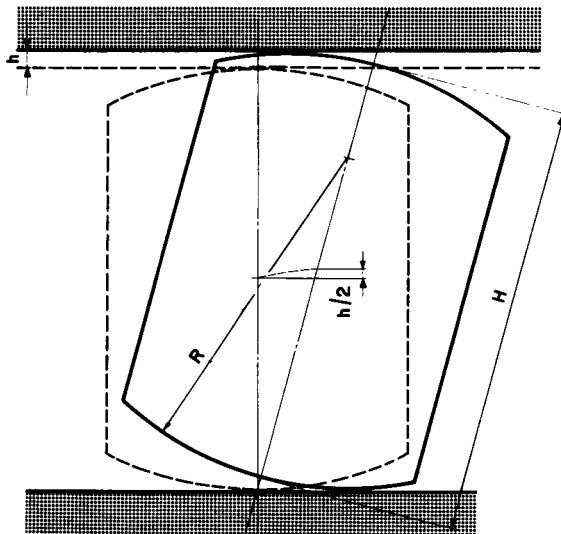


FIG. 7. Operating principle of rocking support.

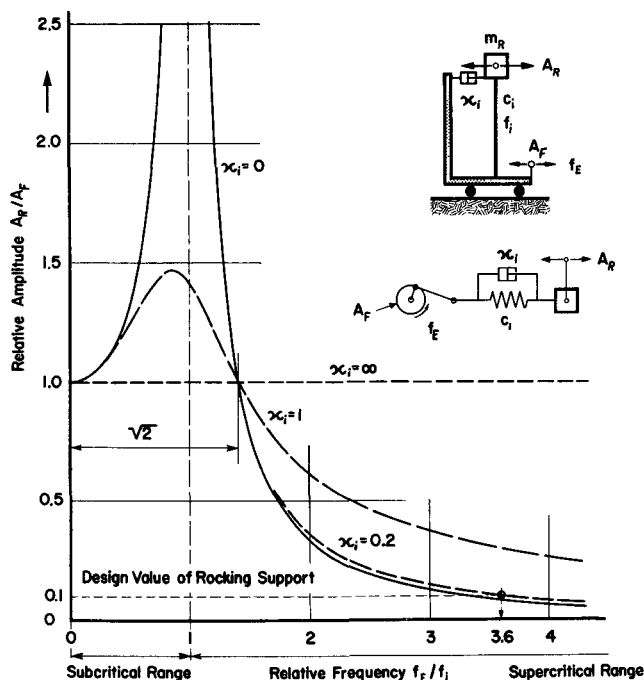


FIG. 8. Resonance curves.

(b) A harmonic oscillation independent of time is assumed to be the exciting function. The results, therefore, show the steady-state condition.

(c) The maximum earthquake acceleration of 0.12 g is assumed to be constant for the reference frequency range. This corresponds to a variable amplitude of 0.3 mm at 10 Hz or 3.3 mm at 3 Hz.

Figure 8 shows the resulting resonance curves. It can be seen that in the super-critical range the building amplitude A_R becomes considerably smaller than the foundation amplitude A_F , if the natural frequency f_1 is sufficiently lower than the excitation frequency f_E . It is, therefore, possible to reduce the maximum amplitude of the reactor mass m_R to a small fraction of the excitation amplitude A_F by tuning the natural frequency of the rocking support system to a correspondingly low value. This constitutes the principle of dynamic protection against earthquakes.

The last two assumptions used for the analysis interpret the reference earthquake pessimistically. Therefore, considerably smaller accelerations will occur at the reactor in reality.

Figure 8 shows that for dynamic protection in the super-critical range any damping worsens the intended protective effect. This also applies if the spring element itself has inherent damping, e. g. internal friction.

Based on the constant coefficient for rolling friction $\mu = 0.01$, and the frequency ratio $f_E/f_1 = 3.6$, which corresponds to an amplitude ratio $A_R/A_F \approx 0.1$; an equivalent damping factor of $\xi_2 = 0.2$ is obtained for the rocking support if equal braking energy is assumed.

This value is very low and hardly reduces the protective effect. Despite this the rolling friction is sufficient to suppress undesired oscillations, for example as a result of Alaska-earthquakes ($a_E \approx 0.001 g$ at 0.1 to 0.2 Hz).

3.4. The rocking support: design and operating principle

The design of the rocking support is shown in Fig. 9. Basically it comprises a column with spherical calottes. They roll freely on plain supporting plates. Tie rods are only necessary for fixing and adjustment during assembly and erection. The radius of the spherical end-faces is larger than half the height of the column. The rocking support therefore always remains in vertical, stable equilibrium. The kinematics of the movement of the rocking support give the following equation for its natural frequency:

$$f_2 = \frac{\sqrt{g}}{2\pi} \frac{\sqrt{2R - H}}{H}$$

This frequency is independent of the load and is therefore influenced only by the geometry of the support. Compared to conventional earthquake protection systems (rubber supporting structures, helical springs) this is of great advantage, especially with variable loads (silos) and for uneven load distribution.

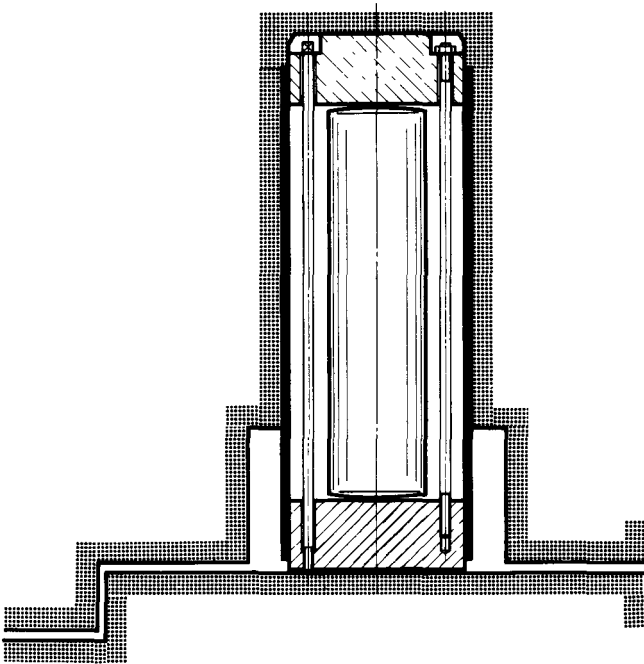


FIG. 9. Layout principle of rocking support.

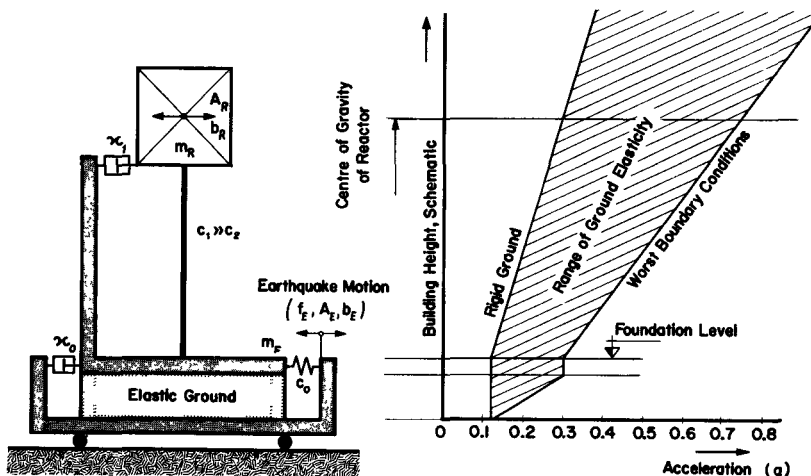


FIG. 6(a). Mathematical model without rocking support. FIG. 6(b). Corresponding acceleration diagram.

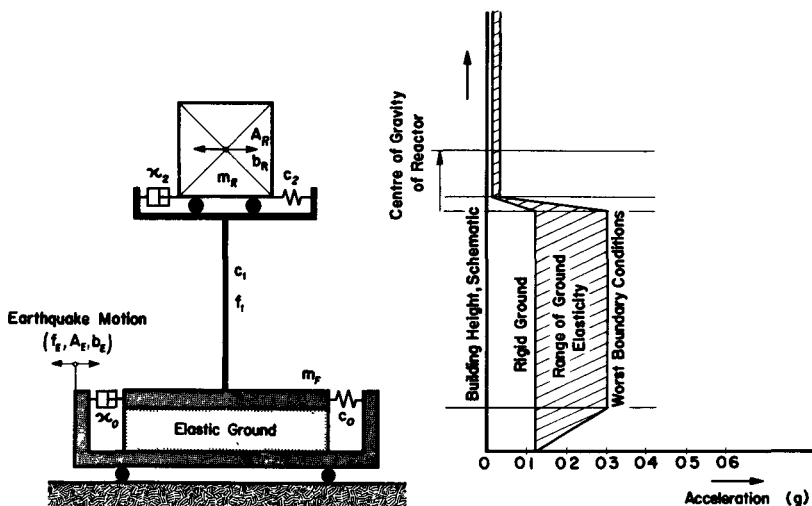


FIG. 10(a). Mathematical model with rocking support. FIG. 10(b). Corresponding acceleration diagram.

Load and permissible surface pressure (Hertzian Pressure) determine the radius R . The desired natural frequency f_2 of the rocking support is then obtained through appropriate dimensioning of its height H .

In our practical example an amplitude ratio of $A_R/A_F = 0.1$ was desired for an earthquake frequency of $f_E = 3$ Hz. According to Fig. 8 this requires a frequency ratio of $f_E/f_2 = 3.6$ which results in a natural frequency of $f_2 = 0.8$ Hz for the rocking support. Material and load conditions dictated a radius of $R = 400$ mm, resulting in a height of 400 mm.

Even with resonance between ground and foundation the amplitude at the rocking support lower plate would remain below 1 mm; the rocking

By locating the rocking support with its 'soft' oscillation response between the top of the foundation wall and the pressure vessel (see Fig. 5) the foundation wall can no longer act as a spring, which results in a constant acceleration over its height. The corresponding mathematical model with rocking support is shown in Fig.10(a). The resultant effect can be seen in Fig. 10(b). The same pessimistic assumptions for ground conditions and foundation embedment were made as for the previous model according to Fig. 6(b). Compared to the unprotected system the considerable reduction in the acceleration transmitted to the building is quite obvious.

The envelope curve F of the resonance maxima represents all resonance cases of the system 'foundation-embedment'. Its value is approximately 0.3 g, which is 2.5 times higher than the earthquake acceleration 'E'.

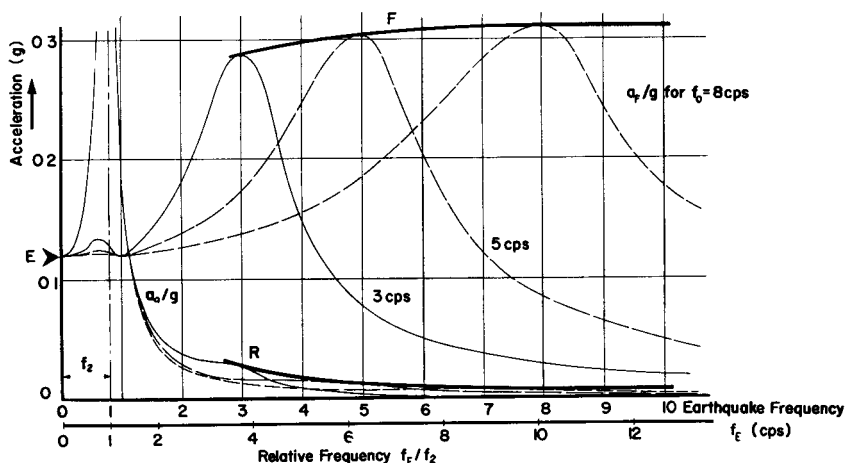


FIG. 11. Acceleration of foundation and reactor (with rocking support) as a function of the relative frequency f_E/f_2 .

The corresponding envelope curve R of the lower family of curves a_R/g joins the maximum accelerations of the building. The rocking support therefore reduces the foundation acceleration from approximately 0.3 g down to less than 0.03 g at the building.

In the reference frequency range the two envelope curves F and R have a flat character, so that the building acceleration is always about 0.1 times the foundation acceleration. This means that shift of the resonance frequency of the building foundation due to time-dependent change of ground conditions, inaccuracies in determining the natural frequency of the foundation or different sites for the same project have practically no influence on the favourable effect of the rocking support.

4. PRACTICAL APPLICATION OF THE ROCKING SUPPORT

The analysis assumed that the rocking support stands vertically in its rest position and both roll plates remain plane, parallel and horizontal. In practice these requirements would be very difficult to achieve owing to the building procedure and the normal tolerances in the building trade. For example, it is possible that concrete creep or shrinkage or differential settling of the ground could tilt the roll plates or the rocking supports to some extent. Such positional errors could also occur after completion of the buildings, caused, for instance, by temperature differences or wind forces. Deviation from the assumed ideal case changes the natural frequency of the rocking support. Calculations carried out for such changes show that these either improve the beneficial effect or can be compensated by appropriate dimensioning of the rocking support. Therefore no unfavourable effects exist and normal building practices can be applied.

Earthquake protection by conventional methods – static or dynamic – is commonly applied and has been used for the design of various nuclear power plants.

Earthquake protection using rocking supports is not limited to nuclear power plants. It can supplement or replace conventional methods in all cases where its application is technically or economically superior.

In conclusion, there are certainly many interesting applications for this method.

R E F E R E N C E S

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LA SURETE DES COMBUSTIBLES NUCLEAIRES Essais effectués dans la pile PEGASE

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Abstract—Résumé—Аннотация—Resumen

SAFETY OF NUCLEAR FUELS: TESTS CARRIED OUT IN THE PILE PEGASE.

The safety experiments carried out in the nuclear fuel testing pile PEGASE can be divided into two categories: fuel resistance tests under abnormal conditions, to define the limits of use of this fuel and, consequently, to increase the yield of power piles without prejudice to their safety; tests aimed at improving the conditions of detection of events liable to compromise seriously the working condition of a power reactor. Tests in the former category have confirmed the very good behaviour of CHINON 3 type cartridges at a cladding temperature of 575°C over a period of more than 3000 hours. The second category included tests on cartridges with calibrated leaks, the aim being to study, under different operational conditions, the behaviour of tubular and graphite-core fuels as a function of the leak size. These tests demonstrated in particular the advantage of reducing the graphite pore size to abolish extensive detachments of the cladding. Two cladding fusion tests were also carried out in this category, one on a BUGEY type annular element and the other on an element of the British Magnox series, the latter in collaboration with the CEGB, the UKAEA and the CEA. It led to the definition of a system of rapid detection of a serious accident occurring in one of the reactor channels, under conditions where the conventional cladding burst detection method would be too slow.

LA SURETE DES COMBUSTIBLES NUCLEAIRES: ESSAIS EFFECTUES DANS LA PILE PEGASE.

Les expériences de sûreté réalisées dans la pile d'essai de combustibles nucléaires PEGASE peuvent se diviser en deux catégories: les essais de tenue du combustible dans des conditions anormales, en vue de définir les limites d'utilisation de ce combustible et donc d'augmenter le rendement des réacteurs de puissance sans nuire à leur sécurité; les essais destinés à améliorer les conditions de détection d'événements risquant de compromettre gravement les conditions de fonctionnement ou d'exploitation d'un réacteur de puissance. Les essais de la première catégorie ont mis en évidence la très bonne tenue des cartouches de type CHINON 3, à une température de gaine de 575°C pendant plus de 3000 heures. Dans la deuxième catégorie on peut citer les essais de cartouches à fuite calibrée qui avaient pour objectif l'étude, dans différentes conditions de fonctionnement, du comportement des combustibles tubulaires et à âme de graphite, en fonction de la taille de la fuite. Ces essais ont permis en particulier de mettre en évidence l'intérêt qu'il y avait à diminuer la porosité du graphite afin de supprimer les décollements étendus de la gaine. Dans cette même catégorie deux essais de fusion de gaine ont été réalisés, l'un sur élément annulaire du type BUGEY, l'autre sur élément de la filière britannique Magnox. Ce dernier a été réalisé en collaboration avec le CEGB, l'UKAEA et le CEA. Il a conduit à définir un système de détection rapide d'un incident grave intervenant dans un canal du réacteur, dans les conditions où la détection de rupture de gaine classique serait trop lente.

ИСПЫТАНИЯ ЯДЕРНОГО ТОПЛИВА НА БЕЗОПАСНОСТЬ В РЕАКТОРЕ PEGASE.

Эксперименты по определению безопасности тепловыделяющих элементов в реакторе PEGASE для испытания ядерного топлива можно разделить на две категории: испытания стойкости топлива в аварийных условиях для определения пределов использования такого топлива и, следовательно, для оценки возможности повышения эффективности энергетических реакторов без нарушения их безопасности; испытания, имеющие целью улучшить условия обнаружения явлений, которые могут подвергнуть реактор серьезному риску. В результате исследований первой категории была выявлена высокая стойкость тепловыделяющих элементов типа "Шинон-III" при температуре оболочки 575°C в течение более 3000 ч.

Ко второй категории относятся исследования тепловыделяющих элементов с калиброванными отверстиями, позволяющие изучить поведение трубчатого топлива и топлива с графитовым сердечником в различных рабочих условиях, в зависимости от величины утечки. Эти исследования позволили, в частности, выявить преимущества применения графита с меньшей пористостью для устранения отслаивания оболочки. Ко второй категории можно также отнести два эксперимента по сплаванию оболочки, один из которых был проведен на кольцевом элементе типа "Бюжей", а другой — на тепловыделяющем элементе британского магноксового реактора. Последние испытания, проведенные в сотрудничестве с Центральным энергетическим управлением Соединенного Королевства, Управлением по атомной энергии Соединенного Королевства и Комиссариатом по атомной энергии Франции, позволили определить систему быстрого аварийного детектирования повреждений в одном из каналов реактора в условиях, когда обычные методы обнаружения разрывов оболочки оказываются слишком неоперативными.

LA SEGURIDAD DE LOS COMBUSTIBLES NUCLEARES: ENSAYOS REALIZADOS EN EL REACTOR PEGASE.

Los experimentos sobre seguridad realizados en el reactor de ensayos de combustibles nucleares, PEGASE, pueden dividirse en dos categorías: ensayos sobre el comportamiento del combustible en condiciones anormales, para definir los límites de utilización de ese combustible y aumentar, en consecuencia, el rendimiento de los reactores de potencia, sin perjudicar su seguridad; ensayos destinados a mejorar las condiciones de detección de sucesos que puedan comprometer gravemente el funcionamiento o explotación de un reactor de potencia. Los ensayos de la primera categoría han permitido constatar el excelente comportamiento de los elementos combustibles de CHINON 3 para una temperatura de la vaina de 575°C durante más de 3000 horas. En la segunda categoría pueden citarse los ensayos con cartuchos de fuga calibrada, que tienen por objeto estudiar, en diferentes condiciones de funcionamiento, el comportamiento de los combustibles tubulares con alma de grafito, en función de la cantidad de la fuga. Estos ensayos han permitido, concretamente, resaltar el interés existente por disminuir la porosidad del grafito para evitar que se produzcan desprendimientos considerables de la vaina. En esta misma categoría se han realizado dos ensayos de fusión de vaina, uno con un elemento anular del tipo BUGEY y el otro con un elemento del tipo británico Magnox. Este último se realizó en colaboración con diversas instituciones nacionales y extranjeras, (CEGB, UKAEA y CEA) y ha permitido definir un sistema para detección rápida de un incidente grave que afecte a un canal del reactor, en condiciones en las que la clásica detección de ruptura de la vaina sería demasiado lenta.

1. LA PILE EXPERIMENTALE PEGASE — CARACTERES SPECIFIQUES

PEGASE est un réacteur piscine refroidi et modéré à l'eau légère, le circuit cœur étant en légère surpression par rapport à la piscine (fig. 1) [1-5].

Répondant à des nécessités expérimentales très spécialisées PEGASE se distingue des autres réacteurs de recherche par les facilités expérimentales particulières qu'il offre. Les irradiations se font dans des dispositifs expérimentaux de grandes dimensions appelés boucles (fig. 2), entièrement autonomes et placés dans le réflecteur autour du cœur à raison de deux par face.

Destinée à essayer des éléments combustibles de réacteur de puissance en vraie grandeur, tant du point de vue géométrique que du point de vue de la neutronique et de l'aérothermique, chaque boucle possède en propre sa motosoufflante, son échangeur de réglage de température du gaz, son circuit émergé d'alimentation et d'analyse de produits de fission et son tableau de contrôle.

Les essais de sûreté d'éléments combustibles, indépendamment des risques d'irradiation qu'ils introduisent, exigent des possibilités particulières de réglage des paramètres et une instrumentation spéciale. La pile PEGASE se prête particulièrement bien à ce genre d'essai par ses «facilités», qui sont essentiellement les suivantes.

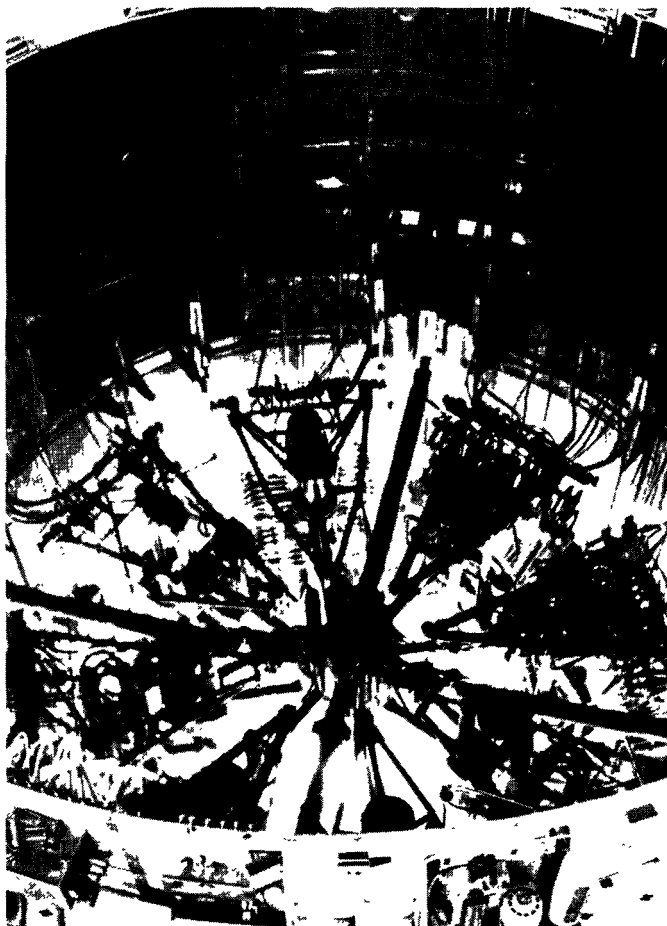


FIG. 1. La pile PEGASE· vue de la piscine et des boucles.

1. 1. Autonomie totale de chaque boucle par rapport au réacteur
et aux autres boucles en irradiation

Une boucle peut être mise en irradiation ou sortie du réacteur sans interrompre ni perturber le fonctionnement des autres boucles.

Ses servitudes électriques, pneumatiques et instrumentales sont particularisées. Chaque poste d'irradiation était à l'origine banalisé, c'est-à-dire qu'une boucle pouvait être irradiée dans un quelconque des huit postes. Du fait de l'évolution des programmes cette propriété s'est estompée et actuellement les postes se sont presque tous spécialisés, sans altérer en rien l'autonomie de chacun d'eux.

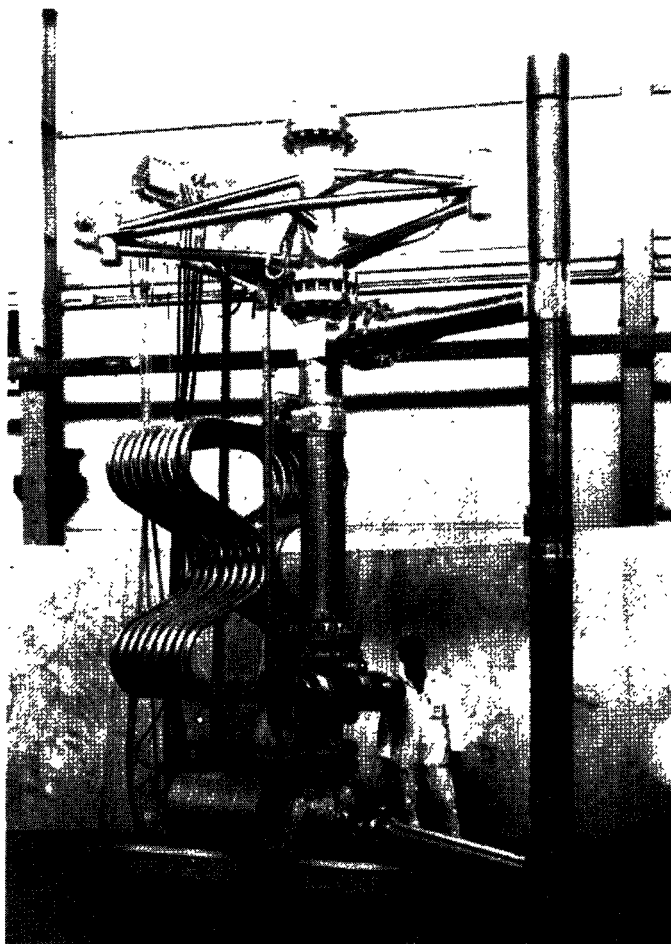


FIG. 2. Boucle INCA.

1. 2. Souplesse et simplicité de réglage des paramètres

Pratiquement toutes les opérations de réglage des paramètres peuvent se faire depuis la salle de contrôle, ces paramètres pouvant varier suivant des lois d'évolution prédéterminées.

a) Le réglage de la température du gaz se fait par action sur une vanne d'échangeur ou, pour certaines boucles, par ajustement de la puissance d'un réchauffeur électrique situé sur la boucle.

b) Le réglage de la puissance thermique se fait par avance ou recul de la boucle par rapport au cœur. Pour ce faire la boucle est posée sur un chariot qui peut se déplacer radialement sur le fond de la piscine. Le chariot est actionné soit manuellement, soit électriquement à partir de la margelle de la piscine.

c) Le réglage de débit de gaz autour de l'élément combustible se fait par changement de fréquence de l'alimentation électrique de la soufflante. Le débit de gaz peut varier dans le rapport 1 à 4.

d) La pression est ajustée par action sur des électrovannes de vidange et de remplissage.

1.3. Fonctionnement aux conditions extranominales [6]

Les boucles, de par leur conception, permettent de fonctionner dans des conditions extranominales, particulièrement pour les boucles de la filière gaz-graphite. La réserve de flux dans ce cas est suffisante pour atteindre des puissances spécifiques nettement supérieures à la puissance nominale. Le nombre important de thermocouples (12 sur l'élément et 6 dans le gaz) permet une surveillance continue et précise des températures.

1.4. Essais d'éléments préalablement irradiés

Des paniers d'irradiation récemment mis au point permettent de charger les boucles en éléments pré-irradiés dans un autre réacteur. Cette technique permet d'utiliser des éléments combustibles provenant de piles de puissance et, par suite, d'économiser sur le temps de séjour dans PEGASE.

1.5. Fonctionnement des boucles en gaz divers

Les boucles, initialement prévues pour fonctionner avec du CO_2 , ont pu être transformées pratiquement sans modifications pour des essais en hélium. Seuls les circuits émergés ont dû être entièrement revus et sont maintenant équipés de circuits d'épuration et d'analyse très complets.

1.6. Radiographie X

Un ensemble de radiographie X installé dans un des bassins de PEGASE permet de visualiser l'élément combustible en essai sans déchargement préalable. L'élément combustible peut être systématiquement radiographié sans interruption notable du temps d'irradiation.

1.7. Boucles existantes

Le parc des boucles de PEGASE est actuellement de 19, dont 18 sont opérationnelles et une en cours de modification:

<u>Dénomination</u>	<u>Nombre</u>	<u>Pression</u>	<u>Diamètre utile</u>
EDF 3	8	25 bars	116 mm
EDF 4	2	30 bars	116 mm
EL 97	6	60 bars	97 mm
EL 150	1	60 bars	150 mm
INCA	2	60 bars	200 mm

Relevons qu'une des boucles EL 97 est spécialement équipée et aménagée pour effectuer des dépressurisations rapides d'éléments combustibles.

1.8. Mesures de produits de fission

Chaque boucle possède en propre son circuit émergé qui est essentiellement constitué d'une détection de rupture de gaine de grande sensibilité spécialement étudiée pour suivre aisément l'évolution d'une rupture de gaine sur le combustible en essai. Des dispositifs nouveaux, par exemple de spectrométrie γ par semi-conducteur Ge-Li, se développent actuellement pour répondre aux exigences des essais HTR en hélium. L'existence d'une salle à atmosphère contrôlée pour l'installation de ces dispositifs simplifie les problèmes d'implantation et atténue les inconvénients des fuites éventuelles de gaz actif.

1.9. Examens post-irradiatoires

La présence d'un laboratoire d'examen des combustibles irradiés à proximité de la pile PEGASE facilite l'examen des combustibles dans des délais réduits après la sortie de pile.

Toutes ces possibilités, jointes à l'expérience de huit années d'un personnel stable et particulièrement bien entraîné, font de PEGASE un outil idéal pour exécuter des essais de sûreté qui présenteraient des risques potentiels inacceptables dans d'autres installations et en particulier dans des réacteurs de puissance [7].

2. ESSAIS DE SURETE SUR ELEMENTS COMBUSTIBLES DE LA FILIERE URANIUM NATUREL GRAPHITE-GAZ

2.1. Objectif

Ces essais peuvent se diviser en deux catégories ayant chacune un objectif différent:

Première catégorie

Essais de cartouches normales à des conditions très sévères de température et de puissance, qui permettent, selon les niveaux atteints, soit de définir de nouvelles limites d'utilisation des cartouches, soit de connaître les dommages que peuvent provoquer certains incidents de pilotage des réacteurs.

Deuxième catégorie

Essais de cartouches à fuite calibrée qui ont pour objet:

- de connaître les possibilités de détection précoce d'un défaut de taille initiale connue dans les différentes périodes du séjour en pile de la cartouche,
- d'établir la cinétique de dégradation de la cartouche,
- d'évaluer le risque pour le réacteur.



FIG. 3. Cartouche du type CHINON 3 après 3200 h de fonctionnement à une température de gaine de 570°C.

2.2. Description des essais de première catégorie

Ces essais ont été effectués sur des éléments combustibles tubulaires de type CHINON 3 [8], en deux séries. Dans la première série l'irradiation a eu lieu à une puissance double de la puissance nominale prévue pour la cartouche en conservant une température de gaine égale à la température maximale admissible; dans la deuxième série d'essais la température de la gaine a été portée très au-dessus de la valeur autorisée avec une puissance normale (simulation d'un accident de baisse de débit du canal). Un des essais de cette série est décrit ci-après.

Essai de cartouche CHINON 3 à haute température de gaine

Cette cartouche a été irradiée dans des conditions de température et de puissance variables:

6,0 W/g	< puissance spécifique moyenne	< 7,30 W/g
530°C	< température de gaine maximale	< 575°C
660°C	< température de l'uranium maximale	< 730°C
débit: 2,2 kg/s		
nombre de cyclages de la puissance maximale à 0 : 104		
durée de l'essai: 3200 h.		

L'examen de la cartouche a mis en évidence son excellent comportement pour la durée de l'essai. La diminution de diamètre du tube combustible (1,5 mm) est acceptable et le fluage des ailettes visible sur la figure 3 ne diminue pas sensiblement les qualités aérothermiques de la gaine.

Cet essai a permis de démontrer qu'une telle élévation accidentelle de température dans le réacteur ne compromettrait pas la limite technologique d'utilisation des cartouches.

2.3. Description des essais de cartouches à fuites calibrées

De très nombreux essais ont eu lieu avec des fuites de taille différente sur les trois types d'éléments combustibles de la filière uranium naturel-graphite-gaz: tubulaire, à âme de graphite et annulaire.

Chaque essai était destiné à reproduire les différentes conditions de fonctionnement de la cartouche pendant son séjour dans le réacteur: chargement, fonctionnement en palier, montée en puissance.

Afin d'illustrer les facilités qu'offre PEGASE pour simuler les diverses conditions de fonctionnement de la cartouche nous décrivons ci-dessous le programme type destiné à l'étude du comportement d'une cartouche à fuite pendant la montée en puissance du réacteur:

a) Montée à 400°C, gaine à faible puissance (simulation du chargement de la cartouche dans le réacteur). La cartouche est conservée pendant 6 h sous 25 bars de pression de CO₂ entre 80 et 100°C. La montée en puissance se fait en dix minutes avec le débit le plus faible possible, la vanne de l'échangeur complètement fermée et le réchauffeur électrique en marche.

b) Trois montées à 5°/min jusqu'au point nominal de CHINON 3 après des temps d'immersion à froid dans le CO₂ sous 25 bars de 1 h 30, 6 h, 24 h. Maintien pendant 8 h au point nominal de CHINON 3.



FIG.4. Essai de fusion de gaine sur élément annulaire.

c) Test de respiration de la cartouche par baisse continue de pression à 0,1 bar/min à une température de gaine de 400°C.

d) Test de respiration par augmentation de puissance de 5% par minute à une température de gaine constante.

Ce programme a permis de mettre en évidence l'influence de la porosité du graphite sur le décollement de la gaine à la montée en puissance après un arrêt de quelques heures. L'examen après irradiation a montré qu'en présence d'un noyau à forte porosité une surchauffe de l'uranium due à la quantité de CO₂ absorbé pendant le séjour de la cartouche hors flux avait lieu.

Dans le cas d'un noyau à faible porosité ces dommages disparaissent totalement. Dans les deux cas d'ailleurs le système DRG à double détection de PEGASE a permis de démontrer que la détection précoce d'un tel incident dans un réacteur de puissance était pratiquement certaine et que la probabilité de provoquer des dommages sérieux dans le combustible était très faible. Ces essais ont néanmoins prouvé qu'il était préférable de diminuer la porosité du noyau pour atteindre une sécurité totale.

3. ESSAIS DE FUSION DE GAINES - SIMULATION D'UN ACCIDENT PROVOQUE PAR UNE OBSTRUCTION GRAVE D'UN CANAL

3.1. Essai sur élément annulaire

Ce type d'essai n'appartient pas à proprement parler aux essais de sûreté qui servent à qualifier un élément combustible. En effet, dans le cas d'une diminution importante du débit de refroidissement la conception de la cartouche ne joue pratiquement aucun rôle; la prévention d'un accident grave est essentiellement liée aux performances du dispositif de DRG dont est équipé le réacteur.

Un essai de fusion de gaine a été réalisé dans PEGASE sur un élément annulaire de 54 × 70 mm équipé de dispositifs de mesure de la température dans la gaine et dans l'uranium.

La montée en température entre 130°C et la température de fusion des gaines s'est faite en 3 min, la vitesse initiale de montée en température a été de 310°C/min. La cartouche a été maintenue pendant 20 min à une puissance spécifique de 6 W/g et à une température maximale de gaine de l'ordre de 560°C.

Le système de DRG de PEGASE ainsi que le compteur à BF₃ qui avait été installé à cette occasion pour une mesure par neutrons retardés ont tous indiqué le moment où la fusion de la gaine est apparue. La surface d'uranium mise à nu était d'environ 7 cm². La figure 4 montre la zone fondue de la gaine externe.

3.2. Essais de fusion du Magnox

Cet essai a été réalisé par l'UKAEA, le CEGB et le CEA en collaboration [9] (fig. 5).

Bien que la fusion d'un canal dans le réacteur de type Magnox n° 2, à Chapelcross, en mai 1967, n'ait pas créé de danger du point de vue de la sûreté, les opérations de remise en état durèrent deux années pleines et



FIG. 5. Essai Magnox: a) cartouche avant irradiation, b) cartouche après irradiation, c) tube d'uranium après dégainage.

la pénalité économique fut lourde. Ceci conduisit à un travail considérable en Grande-Bretagne pour déterminer un dispositif faisant tomber les barres du réacteur avant qu'un tel accident puisse se propager d'une manière désastreuse. De tels incidents limités à un seul canal ne proviennent pas de défauts de l'élément combustible mais plutôt d'un mauvais rapport flux/débit de gaz de refroidissement dû à un chargement incorrect ou à la présence de débris, et il est impossible d'éliminer totalement ce genre d'incident.

Des études théoriques montrent que le système de protection approprié doit être capable d'agir rapidement en 20 ou 30 secondes et avec une sûreté suffisante qui évite toute action intempestive. Il en résulte donc que l'activité du gaz caloporteur, due à la mise à nu de l'uranium, doit être mesurée directement avec une grande sensibilité et une grande confiance. Des expériences préliminaires et des analyses ont indiqué que la mesure par neutrons retardés dans le circuit primaire pouvait fournir la base d'un système utilisable.

Pour les centrales à caisson métallique que nous considérons ici (de Calder Hall à Sizewell B), l'existence de tuyauteries de gaz à l'extérieur du caisson et à la partie supérieure du réacteur autorise l'installation d'un tel système de mesure global et évite d'avoir à prélever le gaz. Le détecteur est un grand tube à BF_3 entouré d'un modérateur et protégé extérieurement par un écran qui élimine les neutrons ne provenant pas de la tuyauterie. Equipé d'une alarme en 2 sur 3 ce système assurera la fiabilité exigée.

Pour déterminer le niveau et le temps de transit des neutrons retardés de nombreux essais ont eu lieu. A cette fin, un élément combustible contaminé ou une feuille d'uranium était introduit à l'intérieur d'un réacteur en fonctionnement. Ces essais ont aussi permis d'optimiser la protection des détecteurs au BF_3 . Les mesures ont montré que la perte de signal due au dépôt des précurseurs dans les canaux était négligeable. Des analyses ultérieures et des essais de laboratoire montrèrent que la présence de produits de combustion ne modifiait pas cette position d'une manière significative.

Cependant, avant de décider l'installation de ces moyens de protection de réacteurs Magnox, deux problèmes essentiels restaient à élucider. Il fallait d'abord expliciter le phénomène de fusion en fonction des propriétés du Magnox, de la perte de coefficient d'échange et de l'élévation de la température de l'uranium. Ensuite il fallait connaître l'état de la surface de l'uranium pendant la fusion et sa faculté d'émission des précurseurs de neutrons retardés. L'essai décrit ci-dessous et réalisé en collaboration UKAEA/CEGB/CEA fut conçu pour obtenir des informations sur ce phénomène.

Description de l'essai effectué à PEGASE

L'essai fait appel aux facilités particulières de PEGASE. Il a été conduit de la façon suivante:

Des essais préliminaires dans PEGGY (maquette de PEGASE à puissance nulle) ont permis de définir des écrans pour obtenir des conditions de flux bien connues sur l'élément combustible au Magnox à essayer. Le panier support d'élément combustible introduit dans une boucle comportait un élément WAGR de série en guise de chauffelette et,

au-dessus, l'élément Magnox muni de thermocouples pour l'uranium de gaine et de gaz. Le gaz s'écoule de bas en haut. Différents appareils de détection de produits de fission ont été installés sur les circuits émergés. Outre le collecteur électrostatique déjà en place on a disposé;

- 2 chambres à bore ou à fission sur le filtre de la boucle dans la piscine,
- 1 bloc de polythène contenant deux compteurs à BF_3 sur la tuyauterie de prélèvement du gaz de la boucle,
- 1 détecteur Na-I sur la même tuyauterie,
- 1 précipitateur électrostatique Plessey MK 11,
- 1 semi-conducteur au Ge-Li au voisinage du filtre du circuit de prélèvement.

L'irradiation d'une feuille d'uranium de surface connue a permis d'étalonner ces différents détecteurs et une horloge permet de les synchroniser.

Le déroulement de l'essai fut le suivant:

- premier temps: stabilisation de tous les paramètres pour une puissance donnée et un débit de gaz donné;
- deuxième temps: en maintenant la puissance constante, réduction rapide du débit de gaz pour obtenir la fusion;
- troisième temps: chute de barres par dépassement de température dans l'uranium.

Deux essais ont eu lieu:

MO 1. Conditions initiales précédant la réduction de débit de gaz:

Puissance totale du combustible:	75 kW
Température de la gaine Magnox:	534°C
Température de l'uranium Magnox:	550°C
Température du gaz à l'entrée Magnox:	505°C
Débit:	1,3 kg/s.

Le débit de gaz est alors réduit à 0,8 kg/s en 50 s, la fusion intervient 8 min plus tard pour une température de gaine de 635°C environ et la chute de barres a lieu 7 min plus tard. La durée totale de l'essai a été de 16 min.

MO 2. Au cours de cet essai on a essayé d'accélérer le phénomène de fusion par augmentation de la vitesse d'échauffement. Conditions initiales avant fusion:

Puissance totale du combustible:	75 kW
Température de la gaine:	562°C
Température de l'uranium:	580°C
Température du gaz à l'entrée:	530°C
Débit:	1,42 kg/s
Pression:	15,5 bars.

Le débit est réduit à 0,7 kg/s en 55 s. La fusion a lieu 2 min 30 s plus tard et la chute de barres 2 min 40 s après. L'essai a duré 6 min.

Pour chacun des deux essais tous les systèmes de détection ont parfaitement fonctionné, et les combustibles ont été examinés ensuite en cellule chaude.

Conclusion sur cet essai

Les résultats ont été satisfaisants et ont montré que, bien avant que l'uranium n'atteigne des températures élevées, la surface mise à nu donne un signal suffisant pour faire tomber les barres.

Ceci a permis d'installer sur des réacteurs un système de protection qui, dans le cas d'un incident de «melt out», ferait tomber les barres suffisamment vite pour éviter des déformations significatives de l'uranium, et ceci avec un minimum de coût de remise en état.

4. CONCLUSION

Ces quelques exemples d'essais choisis parmi les plus spectaculaires montrent l'intérêt incomparable de la pile PEGASE dans l'exécution d'essais de sûreté d'éléments combustibles de réacteurs refroidis au gaz. La maîtrise de ces essais a été acquise surtout grâce aux qualités particulières d'adaptation du réacteur et de ses boucles. Ces essais permettent d'apprécier la fiabilité des éléments combustibles et, de ce fait, participent à la rentabilisation des installations électronucléaires qui les utilisent.

En effet, le rendement de ces installations peut être augmenté parce que l'on connaît mieux les limites admissibles des paramètres thermiques et mécaniques. En outre, une meilleure appréciation des dangers encourus lors d'un incident sur le combustible permet de différer l'arrêt de l'installation ou tout au moins de programmer cet arrêt, s'il est nécessaire, en limitant les inconvénients d'un délestage brutal du réseau et les risques encourus par choc thermique sur les équipements de l'installation.

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SITING NUCLEAR POWER PLANTS IN THE CZECHOSLOVAK SOCIALIST REPUBLIC

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Abstract-Résumé-Аннотация-Resumen

SITING NUCLEAR POWER PLANTS IN THE CZECHOSLOVAK SOCIALIST REPUBLIC.

Czechoslovakia is greatly interested in developing nuclear power. This interest is based both on the great difficulty in meeting current energy requirements and on the restrictions imposed by conventional resources (fossil fuels, water) for the production of energy. At the same time, Czechoslovakia, which has a high population density and is highly industrialized, would prefer to develop a new source of energy with maximum reliability and safety, which would not increase pollution of the environment through wastes exhausted to the atmosphere. In all phases of design, construction and operation of nuclear power plant, the norms, standards and regulations issued by the Ministry of Health and the Czechoslovak Atomic Energy Commission are strictly and carefully followed.

For actual siting, two different approaches are used: siting in relatively isolated locations, i.e. nuclear power plants without containment, surrounded by exclusion areas, with maximum emphasis on reliability of all components (power plants A-1, VVER-440); siting nuclear power plants (particularly for heating purposes) directly in highly populated areas with a high consumption of electricity and heat. This approach demands the use of a well-proven type of reactor with double containment and all engineered safeguards (studies of nuclear heating plants in Prague and Brno).

In both cases, careful safety analyses are done, both for accident situations and for normal operation of nuclear power plants. On the basis of these studies, suitable design criteria are chosen for the future economical and safe development of nuclear power in Czechoslovakia.

IMPLANTATION DES CENTRALES NUCLEAIRES DE PUISSANCE DANS LA REPUBLIQUE SOCIALISTE TCHECOSLOVAQUE.

La Tchécoslovaquie s'intéresse très sérieusement au développement de l'énergie nucléaire. Cet intérêt est fondé à la fois sur un équilibre énergétique très précaire et sur les possibilités limitées de production d'énergie à partir des ressources traditionnelles (combustibles fossiles, houille blanche). Parallèlement, la Tchécoslovaquie, qui a une forte densité de population et qui est très industrialisée, préférerait développer une nouvelle source d'énergie offrant un maximum de sécurité et un minimum de risque et qui n'augmenterait pas la pollution de l'environnement par le rejet de déchets dans l'atmosphère. A tous les stades de l'étude, de la construction et de l'exploitation des centrales nucléaires, les normes, mesures et règlements émis par le Ministère de la Santé et la Commission tchécoslovaque de l'énergie atomique sont scrupuleusement suivis.

Selon le site, on emploie deux systèmes différents: Dans des endroits relativement isolés, centrale nucléaire sans enveloppe de sécurité, entourée de zones d'exclusion. On insiste au maximum sur la fiabilité de tous les composants (centrales A.1, VVER-440). Dans les régions à forte densité de population où la consommation d'électricité et de chaleur est grande, centrale nucléaire (particulièrement pour la production de chaleur) d'un type éprouvé avec une double enveloppe de sécurité et toutes les protections incorporées (étude des centrales nucléaires de Prague et Brno).

Dans les deux cas, des analyses approfondies de la sûreté, portant aussi bien sur des situations accidentelles que sur l'utilisation normale, sont faites. Sur la base de ces études, on détermine les critères appropriés pour l'étude du développement futur économique et sans danger de l'énergie nucléaire en Tchécoslovaquie.

ВЫБОР ПЛОЩАДОК ДЛЯ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ В ЧЕХОСЛОВАЦКОЙ СОЦИАЛИСТИЧЕСКОЙ РЕСПУБЛИКЕ.

Чехословакия исключительно заинтересована в развитии атомной энергетики. Этот интерес вызван очень напряженным энергетическим балансом и ограниченными возможностями производства энергии за счет классических источников (ископаемые виды топлива, гидроэнергия). Одновременно Чехословакия как страна с очень высокой плотностью населения и развитой промышленностью стремится использовать новый энергетический источник с максимальной надежностью и безопасностью, так чтобы не происходило загрязнения окружающей среды. На всех этапах проектирования, строительства и эксплуатации атомных электростанций строго и тщательно соблюдаются нормы, критерии и рекомендации, изданные Министерством здравоохранения и Чехословацкой комиссией по атомной энергии.

В настоящее время для выбора площадок используются два разных подхода: месторасположение атомных электростанций в относительно изолированных местностях, т.е. атомных электростанций без противоаварийных оболочек, окруженных санитарно-защитными зонами, при максимальных требованиях обеспечения надежности всех компонентов (атомные электростанции А-1, VVER-440); месторасположение атомных электростанций (особенно для производства тепла) непосредственно в районах с высокой плотностью населения и с большой потребностью в электроэнергии и теплоте. Этот подход требует использования реактора проверенного типа с двойной оболочкой и проведение всех остальных технических мероприятий по безопасности (проекты АТЭЦ в Праге и Брно).

В обоих случаях проводится тщательный анализ безопасности для аварийных ситуаций и для нормальной эксплуатации атомных электростанций. На основании результатов этих анализов выбираются подходящие критерии для экономического и безопасного развития атомной энергетики в Чехословакии.

UBICACION DE CENTRALES NUCLEARES EN LA REPUBLICA SOCIALISTA CHECOSLOVACA.

Checoslovaquia está vivamente interesada en el desarrollo de la energía nucleoelectrónica. Este interés se basa a la vez en su gran dificultad para satisfacer las necesidades energéticas actuales y en las limitaciones de la producción de energía mediante los recursos convencionales (combustibles fósiles, recursos hidráulicos). Al mismo tiempo, Checoslovaquia, que tiene una elevada densidad de población y está muy industrializada, prefiere desarrollar una nueva fuente energética con un máximo de fiabilidad y seguridad, sin aumentar la contaminación del medio ambiente debida a los desechos arrojados a la atmósfera. En todas las fases del diseño, construcción y operación de las centrales nucleares se observan estricta y cuidadosamente las normas, disposiciones y reglamentos establecidos por el Ministerio de Sanidad y por la Comisión de Energía Atómica Checoslovaca.

Para la ubicación de una central nucleoelectrónica se siguen dos criterios diferentes: en lugares relativamente aislados se instalan centrales nucleares sin contención, rodeadas por zonas despobladas, dándose la máxima importancia a la fiabilidad de todos los componentes (centrales nucleares А-1, VVER-440); en áreas muy pobladas, con gran consumo de electricidad y de calor, se instalan centrales nucleares (especialmente con fines caloríficos) con un tipo garantizado de reactor con doble contención y todas las garantías de ingeniería estructural necesarias (estudios de centrales nucleares para calefacción en Praga y Brno).

En ambos casos se realizan estudios muy completos de las condiciones de seguridad tanto para casos accidentales como para el funcionamiento normal de las centrales nucleares. Basándose en esos estudios se han seleccionado adecuados principios de diseño para el futuro desarrollo, económico y seguro, de la energía nuclear en Checoslovaquia.

1. INTRODUCTION

The Czechoslovak Socialist Republic has a special interest in the development of nuclear power. This interest is due, not only to the unfavourable power situation in the country at present, but also to the limitations imposed by conventional sources, i.e. fossil fuel and hydro-electric plant. For this reason a new construction program for nuclear

power plants has been assessed, which should be able to cope with the growth of power consumption in the coming years. A decision has been made to co-operate with the USSR in setting up two nuclear power plants of the VVER 440 type with a total capacity of 1700 MW(e) in the next five years. By 1990, pressurized-light-water reactors with a total capacity of 12 000 MW(e) in nuclear power are to be installed in Czechoslovakia [1].

2. NATURAL CONDITIONS FOR SITING NUCLEAR POWER PLANTS

Czechoslovakia is a small country in the centre of Europe with certain specific natural conditions, among which the density and distribution of the population and the shortage of surface water are decisive for nuclear power plant siting.

The high population density in Czechoslovakia can be illustrated as follows:

	inhabitants/km ²
World average	20
European average	55
German Democratic Republic ¹	158
Czechoslovakia	110
Hungary	107
Bulgaria	70
United States of America	20
USSR	10

The population is relatively evenly distributed and is not concentrated in large centres. Czechoslovakia has only four cities with more than 200 000 inhabitants (only Prague has over 1 million) and 120 communities have 10 000 inhabitants and more.

The average distance between communities is very short – about 2 km. The population is less dense in mountainous areas and in the less industrialized regions, namely in the southern and western part of Bohemia, in the Czech-Moravian Hills and the Jeseníky Mountains, and in the central and eastern parts of Slovakia.

The rivers in Czechoslovakia are the main source of utility water for the towns and cities, for industry and for agriculture. In many cases they are also an important source of drinking water. Czechoslovakia has no powerful river flows apart from the Danube (Table I). In view of the fact that the national economy needs ever-increasing quantities of water, Czechoslovakia faces the necessity of economizing on this natural resource. With regard to thermal pollution, the Czechoslovak Government Decree No.74/56 of 1954 permits a change of ± 5 degC from the natural temperature;

¹ The designation of countries or territories and the arrangement of material in this publication should not be considered as implying any endorsement or other judgement by the United Nations or the International Atomic Energy Agency regarding the legal status of any country or territory, or of its authorities, or in respect of the delimitation of its boundaries.

TABLE I. AVERAGE FLOWS OF MAIN RIVERS IN THE PERIOD 1901 - 1960

River (measuring station)	Average water flow (m ³ /s)	Monthly minimum in percentage of annual average
Elbe (Brandýs nad Labem)	100.3	57.4
Vltava (Modřany)	145.3	66.1
Ohře (Louny)	37.0	50.0
Elbe (Děčín)	305.7	61.5
Odra (Bohumín) (1926 - 1960)	42.2	70.1
Dyje (Dolní Věstonice) (1936 - 1960)	43.8	61.5
Morava (Moravský Ján)	113.9	55.8
Danube (Bratislava)	2022	72.2
Váh (Trnovec nad Váhom)	157.9	68.2

in the summer months the temperature of trout water must not exceed 20°C, while that of non-trout water must not exceed 25°C.

The specific characteristics of Czechoslovakia – high population density, short average distance between communities, and problems related to the water supply – require that the regulations and rules on radiation protection issued by the Ministry of Health and by the Czechoslovak Atomic Energy Commission be strictly observed. These regulations consistently follow the recommendations of the International Commission for Radiation Protection. For protection of the population, the Czechoslovak regulations implement the assumption of the linear dose-effect relation. From this point of view, doses are considered permissible for the population in the vicinity of an installation which do not exceed 1/10 of permissible doses for occupational personnel. Such doses are permissible for individuals or for a critical group of the population. For large groups of the population, the lowest attainable doses of ionizing radiation are stipulated, and in all cases it is necessary to assess the benefit derived from nuclear power against the possible risk of late effects of ionizing radiation on the population.

On the basis of the above, the choice of sites in Czechoslovakia for the construction of nuclear power plants requiring wide protective (exclusion) zones is difficult and limited both geographically and economically. The

TABLE II. EXCLUDING CRITERIA FOR SITING

<u>Geological conditions</u>
Areas with landslides
Karst areas
Areas that are undermined at present or that will be in the future
Areas with deposits of raw materials
Areas with gradient exceeding 10%
Areas with seismicity exceeding VIII° MCS
<u>Water protection conditions</u>
Protection of important (prospective) sources of drinking water
<u>Conditions for nature conservation</u>
Spa areas
National parks and large reservations
Areas intensively used or intended for use for recreation, tourism and hunting
<u>Conditions of agricultural production</u>
Areas where arable soil should be protected
<u>Transport conditions</u>
Flight paths and prohibited zones of airfields
<u>Special conditions</u>
Areas suitable for the siting of conventional coal-fired power plants

major part of the densely populated territory of the Czechoslovak Socialist Republic will require nuclear power plants with a high standard of engineered safeguards, including containments.

3. SITING NUCLEAR POWER PLANTS IN AREAS WITH LOW POPULATION DENSITY

Over the past ten years site selection for the nuclear power program has been carried out in Czechoslovakia. The method of processing and evaluating the data acquired on the individual sites was sufficiently flexible to be used for reactors of various types and power requirements with regard to the quality of the environment. The final result of the selection was studied and 40 sites were evaluated for nuclear power plants of the VVER 440 type.

An example of the application of the siting criteria (some of which – the excluding criteria – are given in Table II) is the actual site of the

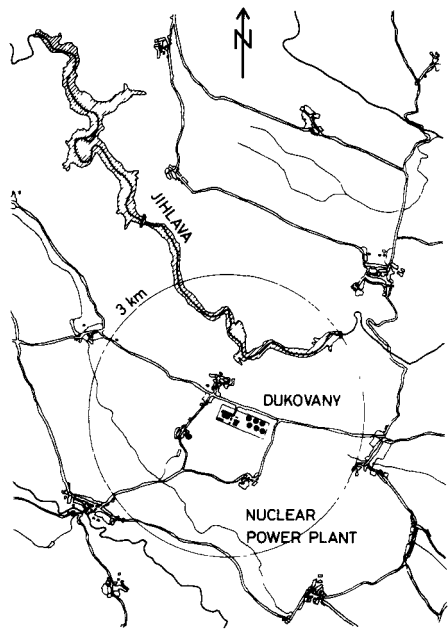


FIG. 1. Nuclear power plant VVER (2 × 440 MW(e)), south Moravia (proposed site).

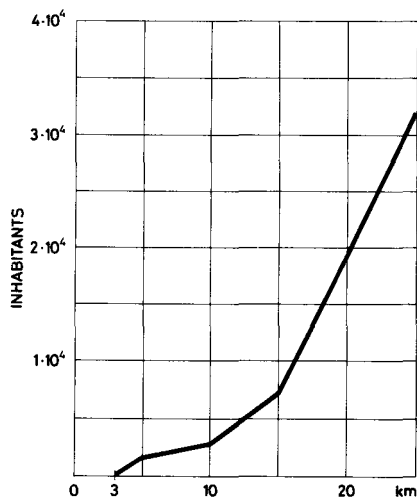


FIG. 2. Nuclear power plant VVER (2 × 440 MW(e)), south Moravia (population distribution around nuclear power plant in N-E sector, 45°).

nuclear power plant VVER 2 × 440 MW(e) in Moravia (Fig.1). The main reasons for the choice of this site were the following:

- favourable demographic conditions (availability of a protective zone, population distribution, etc.);
- the existing adequate supply of water;
- favourable configuration of terrain;
- co-operation between the nuclear power plant and the peak hydroelectric power plant;
- favourable hydrometeorological conditions.

Within the protective zone ($R = 3$ km) there are only three communities with a relatively small number of inhabitants who will have to be moved out.

The size of the protective zone was determined to ensure that the per-annum dose equivalent would not exceed 0.17 rem/yr. An example of a typical distribution of population around the site of a nuclear power plant is given in Fig.2.

The first Czechoslovak nuclear power plant A-1 (natural U, D_2O , CO_2 , critical 1971) and the proposed sites for several other power plants with reactors of the present VVER type are also located in the vicinity of hydroelectric-power plants and in relatively sparsely populated areas.

Nuclear power plant without containment emphasizes the reliability of the technological equipment. Strict requirements are placed on quality control of all important components during manufacturing, commissioning and throughout the lifetime of the power plant.

4. SITING NUCLEAR POWER PLANTS IN AREAS WITH HIGH POPULATION DENSITY

For nuclear power plants or heating plants in areas with a high population density, the siting criteria are considerably reduced and to some extent replaced by technical safety measures. Only exclusion criteria remain valid, which, of course, by their nature also prevent, or have historically prevented, the establishment of big towns and cities. The fundamental safety requirements are the use of a well proven type of reactor and a well designed and reliable system of engineered safeguards.

The design studies of the Brno heating plant and of the Prague power plant serve as examples of urban siting. The growing demands for electrical and thermal energy in industry and for district heating will, it is proposed, be met in Prague by the construction of two nuclear power plants inside this urban agglomeration. The first of the two proposed nuclear power plants to be located in the Holešovice district (Fig.3, A) would meet the needs of the northern and central parts of the city. The second would be designed to supply the new, large housing developments in the south of Prague (the Bráník-Modřany area, marked B in Fig.3). This would greatly help to reduce environmental pollution in Prague. In certain parts of Prague, the fall of fly ash (from the intensive use of low-quality brown coal for heating and as a result of the unfavourable, hilly configuration of the terrain) reaches a value of

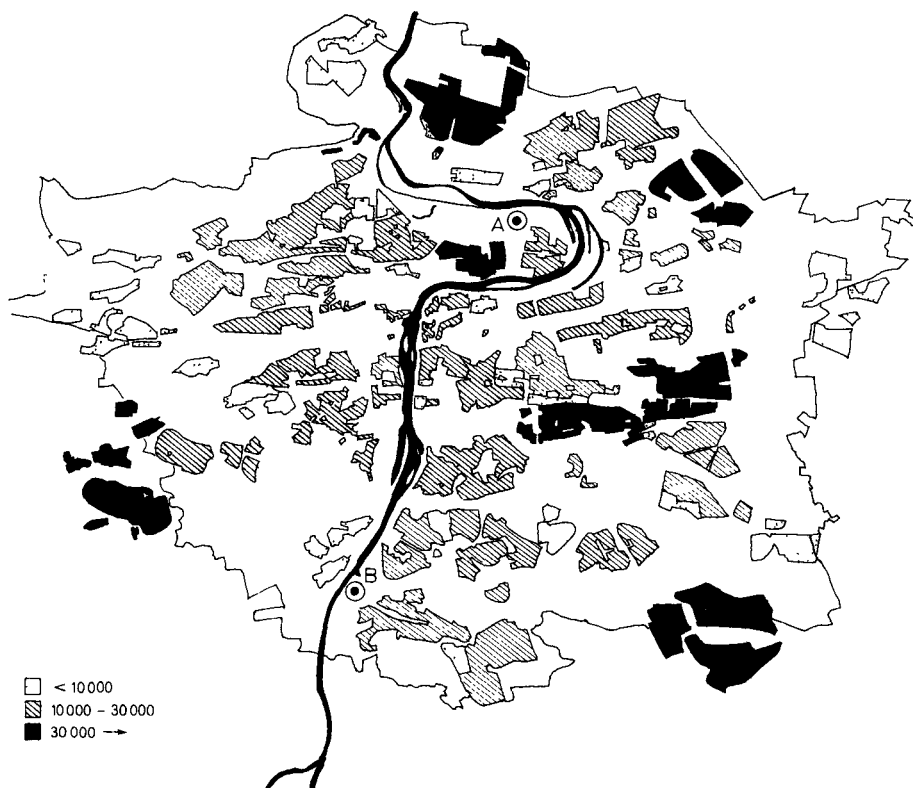


FIG. 3. Population distribution (1980) and proposed sites for two nuclear power plants in Prague (A - location in Holešovice, B - location in Modřany-Bráník).

2000 t/km² yr. An advantage for the proposed construction of nuclear power plants in Prague is the fact that the Vltava river water flowing through the city is considerably undercooled by a system of water dams on the upper flow. This makes it possible to use throughflow cooling with higher power production efficiency.

In accordance with the long-term program for Czechoslovak nuclear power development, the Prague nuclear power plants would each be equipped with two VVER-type reactors with a capacity of 1400 MW(th). The first to be constructed would be the Holešovice nuclear power plant, which would replace an old coal-fired obsolete power plant, the capacity of which no longer meets the demand. The site of the power plant is in the centre of the city (Fig.4), but the river, a large park, and sports grounds in the close vicinity of the power plant will enable a low population zone (up to 500 m) to be established. The city district of Holešovice will be adapted and modernized in the future, so that a power plant of modern design would fit into the environment from the urbanistic and architectural

point of view. Two reactor units are to be built on the relatively restricted site, each unit consisting of a spherical reactor building, an auxiliary operation building and an engine hall. The proposed height of the ventilation stack is 100 m. The construction will proceed gradually with the replacement of the old conventional plant buildings by the new power plant.

The main section of the nuclear power plant will be formed by two nuclear reactors of the VVER 440 type, while for the other section the use of the development series reactor is being considered. The first unit, designed as a heating plant, is equipped with two bleeder steam turbines with full flow condensation, each with a capacity of 220 MW, and one back-pressure turbine with a capacity of 50 MW. It is proposed to equip the second unit with one condensing turbine with a capacity of 500 MW.

The estimated capacity of the power plant is:

Winter	1st unit	500 Gcal/h for heating purposes, approx. 350 MW of electrical power
	2nd unit	500 MW of electrical power
Summer	1st unit	approx. 50 Gcal/h of heat, approx. 435 MW of electrical power
	2nd unit	500 MW of electrical power

The second unit with condenser can also work during breakdowns and refuelling periods as a replacement for the first unit in heat supply.

The Holešovice nuclear power plant (Fig.5 shows a photograph of a model) is an exceptional problem where nuclear safety and the protection of the population living in the neighbourhood of the plant are concerned. Special attention has therefore been paid to the safety of the plant since the very beginning and at all stages of design.

The nuclear reactors are placed in a double containment comprising an outer concrete shell and an inner steel shell with a space between them to enable possible leakage from the inner containment to be controlled. The containment confines radioactive products released in the event of an accident. A number of other engineered safeguards have also been applied in reactor design. In further work, the possibility will also be considered of using an ice condenser containment. For the given conditions of the proposed site (demographic, geographic, meteorological, etc.) a series of environmental calculations have been completed for both normal and accidental releases.

At present, Prague has approximately 1 million inhabitants in the urban area (Fig.3) and approximately 1.3 million with the suburbs. Although the residential districts stretch out on hilly terrain with a difference of altitude of up to 100 m and the power station is located in the river valley, the calculations showed that the proposed nuclear power station would not create an unacceptable risk for its environment. Annual individual doses in Prague would not exceed 1/100 of the doses quoted by the ICRP. For an accidental release, the maximum individual doses have been computed at 0.2 rem and the maximal value of population dose has been computed at 1200 man-rem. The method and results of this evaluation, including figures, are shown in the Annex.

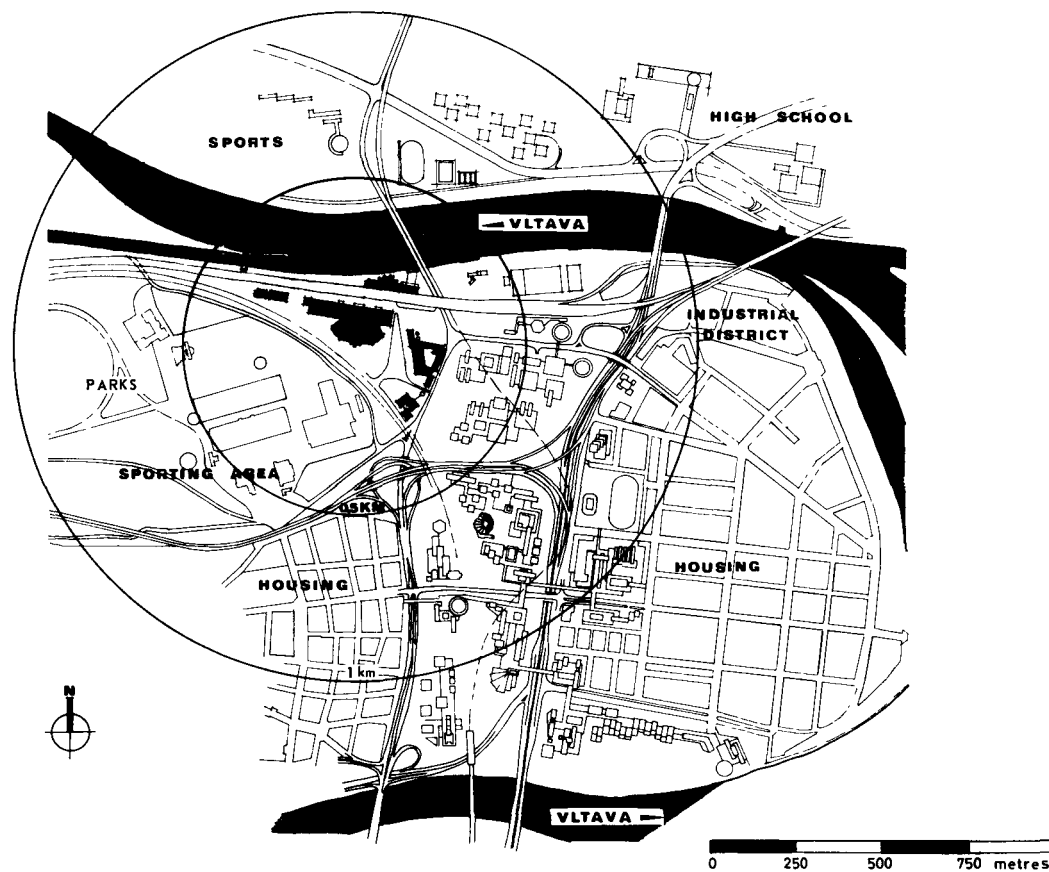


FIG. 4. Nuclear power plant in Prague-Holešovice (proposed site).



FIG. 5. Nuclear power plant in Prague-Holešovice (photograph of a model).

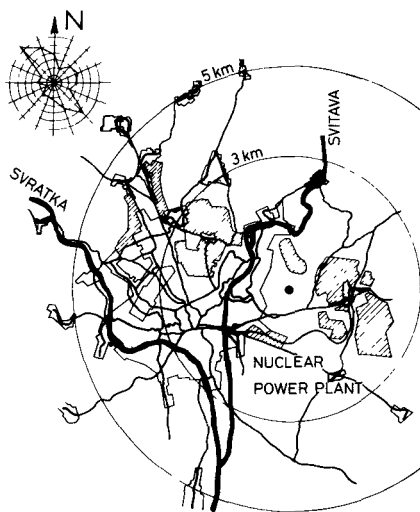


FIG. 6. Proposed site of nuclear heating plant in Brno.

The Brno nuclear heating plant, the design of which is also under study, should supply the district heating system of the city of Brno. It is intended to provide a rated heat supply to the hot water and steam line systems of approximately 370 Gcal/h and it will generate electrical power up to 110 MW(e) (measured on the clamps of the generator). The Brno nuclear heating plant will also be equipped with a VVER-type light-water cooled and moderated reactor, fuelled by slightly enriched uranium.

The proposed site of the plant is to the north-east of the city centre (Fig.6), at the mid-point of current heat consumption and near to a new industrial and housing area that is under construction. The site is also suitable from the climatic point of view because the prevailing winds will disperse the exhaust from the ventilation stack away from the urban area. The height of the stack has been set at 80 m.

5. CONCLUSIONS

Certain natural characteristics of Czechoslovakia influence the siting of nuclear plants. On the one hand, the lack of cooling water orients the selection of sites to places near water reservoirs with pumped storage hydroelectric power plants. These reservoirs are usually located in remote areas suitable for nuclear power plants requiring larger protective zones. On the other hand, the proposed nuclear power plants in Prague, Brno and other areas of high population density require the highest standard of engineered safeguards to ensure the safety of populations.

The first and foremost requirement in both cases is the high quality of the construction of the components of the power station. This require-

ment must consistently be enforced throughout the design, manufacture, assembly, commissioning and operation of the plant. Experience gained so far in nuclear power gives assurance that the construction of nuclear power plants on water reservoirs will not damage the surroundings and that the interference with nature will be limited to the period of construction. The installation of nuclear power plants in towns and cities or replacement of conventional power plants by nuclear power plants can become one of the best ways of reconstruction and improvement of the environment in population centres.

ANNEX

PRELIMINARY SAFETY ANALYSIS FOR THE PRAGUE-HOLEŠOVICE NUCLEAR POWER PLANT

1. INTRODUCTION

The preliminary safety analysis for the nuclear power plant in Prague-Holešovice [2] was carried out in 1970 as part of the technical study of the plant with the aim of analysing important technical and safety aspects of siting a nuclear power plant in an urban agglomeration – in this case in the northern part of Prague. On the basis of this analysis, the overall risk and the effect of individual factors were determined and the basic design requirements were decided upon.

The work was carried out in accordance with the instructions for the organization and content of safety reports [3] and the principles for the safe operation of nuclear reactors [4] drawn up by the Nuclear Safety and Safeguards Department of the Czechoslovak Atomic Energy Commission.

The safety analyses were carried out for (a) a reference accident, and (b) normal nuclear power plant operation.

In the event of an accident, the study was aimed to determine the overall activity dispersion of dangerous radionuclides released from the ventilation stack. In the calculations, attention was mainly devoted to the radioactive isotopes of iodine (^{131}I to ^{135}I) which constitute the greatest risk for the population.

2. PREPARATION OF INPUT DATA

To facilitate the calculation, the distribution of the population of Prague in 1980 for an area of approximately 600 km^2 and the detailed topography of the same area were determined. Both data collections were then processed into a square mesh ($a = 500 \text{ m}$). For the purpose of the calculations, the centres of the square areas were presented by z_{ij} – the altitude above sea level; the total population of the given area was P_{ij} .

No meteorological survey had been carried out at the proposed site and it was therefore necessary to make use of corrected data on the area of Prague obtained from three meteorological stations. The character of the site locality, i.e. the river valley which meanders

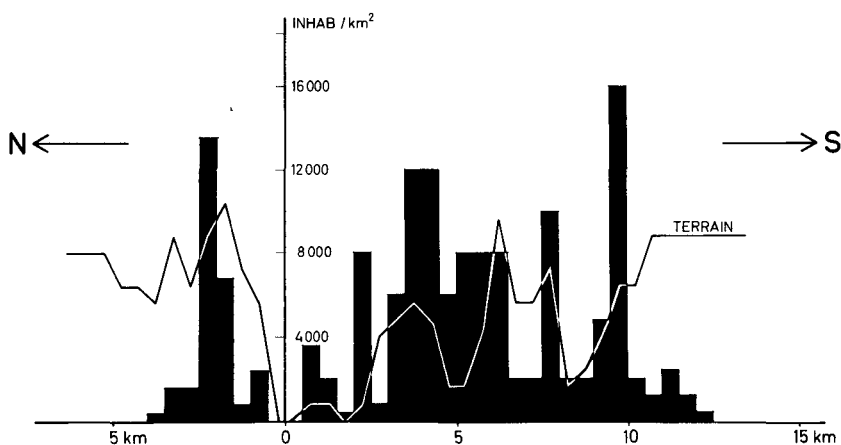


FIG. 7. Nuclear power plant in Prague-Holešovice (population density and ground relief in N-S direction).

between relatively steep slopes, differs considerably from the ideal plain for which the method used of calculating the atmospheric dispersion is valid. The Institute of the Physics of the Atmosphere of the Czechoslovak Academy of Sciences has therefore produced an expert report on local conditions at the selected site [5]. The characteristics of the site in the exposed north-south direction, i.e. the ground relief and the population density per square kilometre, are shown in Fig.7.

3. THE METHOD AND ITS MODIFICATION

The Pasquill and Gifford method was used to determine the individual dose D_{ij} (rem) at the mesh point (i, j) of all iodine isotopes for the weather category W [6]

$$D_{ij}^w = \frac{k_{sft}}{\pi} \sum_{k=131}^{135} \frac{Q_k a_k}{w_{\sigma_y(x)} w_{\sigma_z(x)} \bar{U}_w} \exp \left[-\frac{1}{2} \left(\frac{y^2}{w_{\sigma_y^2(x)}} + \frac{(h-Z)^2}{w_{\sigma_z^2(x)}} \right) + \frac{\lambda_k x}{\bar{U}_w} \right] \quad (1)$$

where

- Q_k is the source strength in Ci/s for the k'th isotope;
- a_k is the conversion coefficient (rem m³/Ci s), according to Ref.[7];
- $\left. \begin{matrix} w_{\sigma_y(x)} \\ w_{\sigma_z(x)} \end{matrix} \right\}$ are horizontal and vertical dispersion coefficients for the spread of radioactive exhaust according to Pasquill;
- $\bar{U}_w$ is the average wind velocity (m/s);
- λ_k is the decay constant (s⁻¹);
- x, y are rectangular co-ordinates of the mesh point (i, j) which, for the pitch diameter of the lattice a and the co-ordinates of the source (i_s, j_s), are

$$x = (i - i_s) \cdot a$$

$$y = (j - j_s) \cdot a$$

h is the actual height of the stack (m);
 Z is a correction allowing for dispersion above an elevated plain (m);
 t is time in hours after steady-state conditions have been reached;
 k_{sf} is the safety coefficient (selected as 2).

If the altitude of the stack base above sea level is denoted as Z_s , then the correction Z has a value of

$$\begin{aligned}
 Z &= 0 & \text{for } Z_{ij} &\leq Z_s \\
 Z &= Z_{ij} - Z_s & \text{for } Z_{ij} - Z_s < 0.8 h \\
 Z &= 0.8 h & \text{for } Z_{ij} - Z_s \geq 0.8 h
 \end{aligned} \tag{2}$$

On the basis of an expert report, characteristic velocities were recommended for individual weather characteristics W:

Weather category W:	A	B	C	D	E	F
Characteristic velocity (m/s):	*	3 - 4	4 - 5	4 - 5	3 - 4	3 - 4

* This category is not applicable to the locality.

In addition to the values of individual doses D_{ij} in densely populated areas, the determination of the population dose value D_{pop} (man-rem) is of considerable significance. In our case, the population doses were determined from the equation

$$D_{pop}^w = \sum_{i_l}^{i_u} \sum_{j_l}^{j_u} D_{ij}^w P_{ij} \tag{3}$$

where $1 \leq i_l, j_l, i_u, j_u \leq 50$.

The values of the upper and lower limits of i_u, j_u and i_l, j_l in Eq. (3) depend on the release co-ordinates in the co-ordinate system (i_s, j_s) and on the angle of wind direction.

The transformation of the co-ordinates at angular rotation φ from the initial direction ($\varphi = 0^\circ$ for the west-east direction) is given by the relationships

$$\begin{aligned}
 x^\varphi &= x^0 \cos \varphi + y^0 \sin \varphi \\
 y^\varphi &= -x^0 \sin \varphi + y^0 \cos \varphi
 \end{aligned} \tag{4}$$

For points with $x^\varphi \leq 0$, $D_{ij}^w = 0$.

All calculations were made for eight main directions, the variable parameters being the height of the stack h (75 - 125 m), weather categories W (B - F) and the average wind velocity $\bar{U}_w$ (3 - 5 m/s).

4. DISPERSION OF RADIOACTIVITY IN THE EVENT OF AN ACCIDENT

A double-end rupture in the main section of the piping was considered as a reference accident. This rupture results in the immediate loss and evaporation of the coolant and the release of iodine from the core.

The safety analysis was carried out as follows:

(1) A preliminary calculation was made of the pressure increase in the steel containment during the accident. The results showed that the maximum overpressure of the air-water mixture reached a value of $\Delta p_{\max} = 4.65 \text{ kg/cm}^2$ in an interval of 7 - 10 seconds. According to the known speed of the closing system and under the justified assumption of the delayed release of the radioactive products, it has been accepted that, in the event of an accident, the uncontrolled leakage of fission products can only result from a leakage in the steel containment.

(2) A complete inventory was made of the fission products. At the end of the fuel cycle in the nuclear reactor chosen, the inventory of iodine isotopes will be:

Isotope	Total Q_a (Ci) inventory
^{131}I	3.83×10^7
^{132}I	5.83×10^7
^{133}I	8.57×10^7
^{134}I	9.68×10^7
^{135}I	7.56×10^7

The actual amount of fission products Q_r released from the ventilation stack after passage through the filters can be expressed as

$$Q_r = k_e k_w k_s (1 - k_f) Q_a \quad (5)$$

where

- k_e is part of the radioiodine released from the total amount after the core damage ($k_e = 0.035 - 0.04$);
- k_w part of the radioiodine absorbed in the cooling water ($k_w = 0.5$);
- k_s part of the radioiodine deposited on various surfaces ($k_s = 0.5$);
- k_f estimated efficiency of filters ($k_f = 0.9$).

With the values of the above-mentioned coefficients, Q_a values decrease by three orders of magnitude according to Eq. (5). For the postulated accident, individual doses were calculated at the mesh point in different directions, and the population doses were calculated from them.

The results of the calculation are shown in Figs 8 and 9. Figure 8 shows the distribution of the individual doses $D_{ij} \geq 0.5 \text{ rem}$ for the stack height $h = 100 \text{ m}$ in the north-south direction; Fig. 9 shows the values of the population doses in all the eight directions for $h = 100 \text{ m}$ with the weather categories B, D, F.

Case A assumes that, during an accident, the containment reactor hall would be constantly ventilated. The calculated dose rates in this

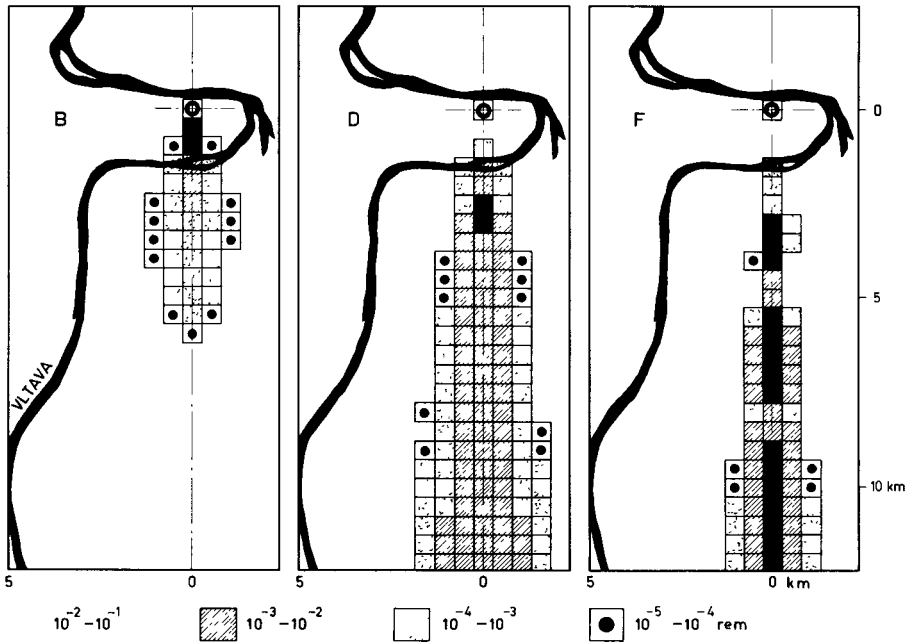


FIG. 8. Nuclear power plant in Prague-Holešovice (calculated individual doses under emergency conditions in N-S direction).

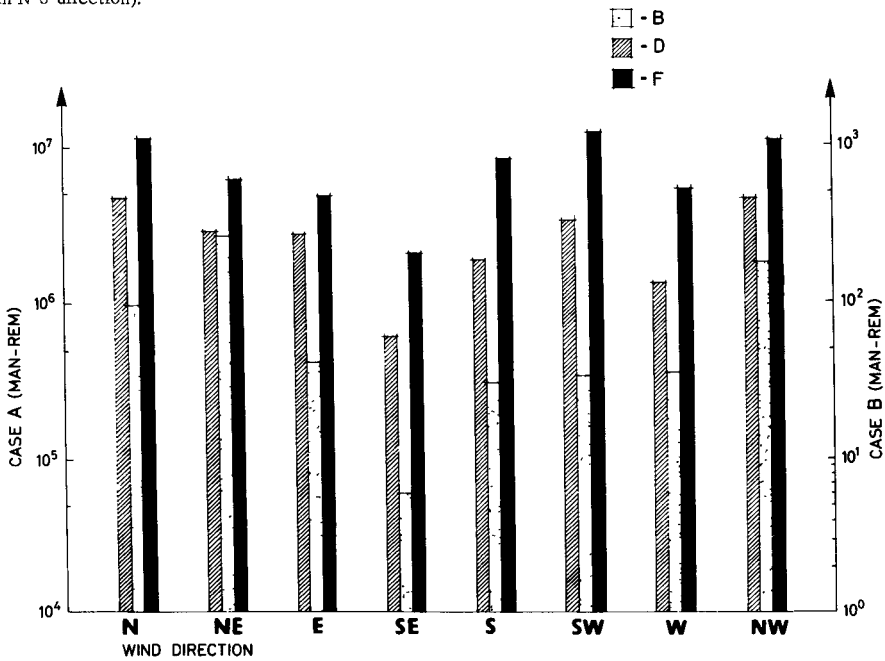


FIG. 9. Nuclear power plant in Prague-Holešovice (calculated population doses under emergency conditions).

case are impermissibly high. In the case of a double containment (case B), the steel containment is closed hermetically during an accident, and only part of the fission products that have leaked into the space between the shells will be released through the ventilation stack. If a leakage rate of 0.01 vol.% per hour is considered, the released activities and, consequently, the calculated doses, are lower by 4 orders of magnitude than in case A. If, in the event of an accident (case B), all engineered safeguards work reliably, according to calculations the individual dose at the mesh points would not exceed 0.2 rem and the population dose would not exceed 1.2×10^3 man-rem. From the point of view of safety, the technical solution involving a double containment is fully acceptable.

The results showed that the height of the stack had a limited effect, and the height $h = 100$ m therefore appears to be sufficient and is also quite satisfactory in other respects.

5. SAFETY ANALYSIS FOR NORMAL OPERATION

The safety analyses must also assess the effect of normal operation of the nuclear power plant on its environment. To determine the risk, the values of doses which the population would receive in the course of one year must be determined. The expert report recommended that the following occurrence of weather categories p_w be taken into consideration:

Weather category W:	A	B	C	D	E	F
Probability of occurrence p_w :	0.00	0.034	0.216	0.573	0.034	0.142

The probability of the direction of the wind p_d is shown in the following table. It clearly indicates that the wind rose is slightly altered for the most stable category F.

Wind direction:	N	NE	E	SE	S	SW	W	NW
Probability of occurrence p_d :								
categories B-E	0.10	0.07	0.13	0.09	0.11	0.16	0.14	0.15
category F	0.10	0.08	0.12	0.15	0.18	0.20	0.10	0.07

The occurrence of calms was evenly distributed in all directions.

With Eq.(1), the total annual dose at point (i, j) is determined by the relationship

$$D_{ij}^{yr} = \sum_{\substack{\text{weather} \\ \text{category}}} \sum_{\substack{\text{wind} \\ \text{direction}}} D_{ij} P_d P_w t_a k_{ys} \quad (6)$$

where

D_{ij}^{yr} is the annual dose at the mesh point (i, j) in rem/yr;
 t_a denotes the number of hours in a year ($t_a = 8.766$ h);
 k_{ys} denotes the conversion factor to determine leakage in Ci/yr ($k_{ys} = 3.169 \times 10^{-8}$).

Overall annual leakage of the respective iodine isotopes, based on design, can be tabulated as follows:

Isotope	Annual leakage (Ci/yr)
^{131}I	45.2
^{132}I	28.4
^{133}I	197.0
^{134}I	10.95
^{135}I	109.5

The results of the calculation of annual doses for $h = 100$ m are shown in Fig.10. The maximum dose ascertained for the entire observed

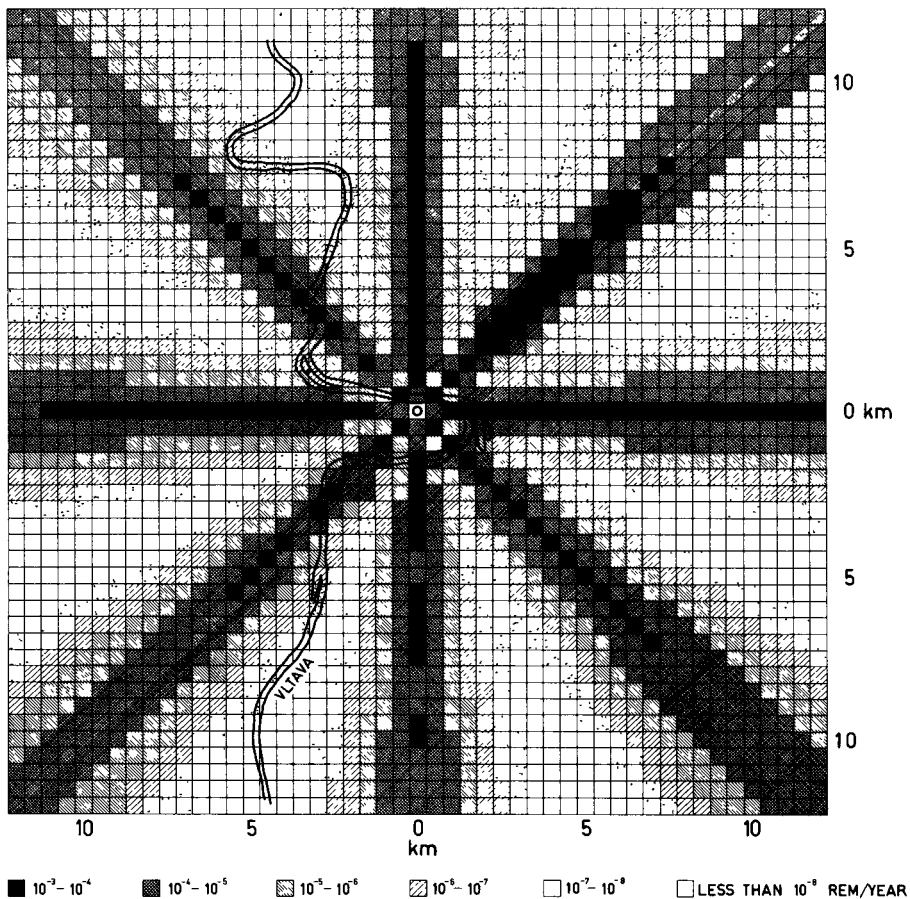


FIG. 10. Nuclear power plant in Prague-Holešovice (calculated annual individual dose distribution).

area equals 1.1 mrem/yr and the average dose for this area is about 0.04 mrem/yr. With a total population of the area of over one million, the annual population dose resulting from the normal operation of the plant is roughly 50 man-rem/yr.

It should be noted that the chosen division of the wind directions in 45° sectors appears to be too crude. If the distinction is finer, the maximum values get "smeared" and decrease by about one order, i.e. to 0.1 mrem/yr.

6. COMPUTER PROCESSING

To enable multi-variant calculations to be done, two versions of the program DOSE were written in the FORTRAN IV code for the IBM computer. The programs will be improved in future on the basis of further development.

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UNDERGROUND SITING OF NUCLEAR POWER PLANTS IN NORWAY

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Abstract-Résumé-Аннотация-Resumen

UNDERGROUND SITING OF NUCLEAR POWER PLANTS IN NORWAY.

An extensive program of investigations on the siting of nuclear power plants underground is being undertaken in Norway. Results so far show that it is technically feasible to site large commercial nuclear power plants in rock caves. The benefits that can be derived from this method of siting are: improved safety and the possibility of urban siting; fewer restrictions on the surroundings; reduced effect on the landscape; and better protection from a civil-defence point of view. Economic assessments indicate that only a marginal cost penalty is associated with this method of siting in comparison with standard siting.

Considerable practical experience with various problems in the underground construction of nuclear power stations has been accumulated in Norway. Thus the 25-MW(th) power test reactor in the city of Halden, the HBWR, is using rock as primary containment and has been in operation since 1959. Most of the new hydro-electric power plants in Norway are also located in rock caves. The research and development program now under way aims at an improved understanding of the ability of different kinds of rock to retain fission products, and at identifying problems connected with the possible contamination of groundwater and with civil engineering limitations in constructing large caverns in different rock structures. Based on the results of this program, further technical and economic assessments of underground nuclear power-plant siting will be made with a view to taking full advantage of the added safety offered by this method of siting.

INSTALLATION SOUTERRAINE DES CENTRALES NUCLEAIRES EN NORVEGE.

La Norvège a entrepris un vaste programme d'études sur l'installation souterraine des centrales nucléaires. Les résultats obtenus montrent que l'installation de centrales nucléaires commerciales dans des cavernes dans la roche est réalisable du point de vue technique. Les avantages qui peuvent être obtenus par ce choix des sites sont les suivants: amélioration de la sûreté et possibilité d'implantation dans les zones urbaines; moins de restrictions dans les zones environnantes; influence réduite sur l'environnement; meilleure protection au point de vue de la défense civile. Les études économiques permettent de constater que la différence entre le coût de ce genre d'installation et celui d'une installation normale est marginale.

La Norvège a acquis une expérience pratique considérable en ce qui concerne les divers problèmes touchant à la construction souterraine d'installations nucléaires. Le réacteur d'essai de Halden (25 MW(th)), réacteur à eau lourde bouillante, est construit dans la roche même qui lui sert de « couche isolante »; il fonctionne depuis 1959. La majorité des nouvelles centrales hydro-électriques de Norvège sont aussi construites dans la roche. Le programme de recherches et de développement actuellement en cours a pour objectif l'étude de la capacité de certaines roches de retenir les produits de fission; il vise aussi à identifier les problèmes relatifs à la contamination des eaux souterraines et à examiner les difficultés et les limitations auxquelles se heurtent les ingénieurs lors des travaux de construction de vastes cavernes dans diverses couches de roches. Les résultats de ce programme permettront de faire une nouvelle évaluation technique et économique du choix des sites de centrales nucléaires pour tirer pleinement profit des avantages de cette nouvelle méthode d'installation en matière de sûreté.

СООРУЖЕНИЕ АТОМНЫХ ЭЛЕКТРОСТАНЦИЙ ПОД ЗЕМЛЕЙ В НОРВЕГИИ.

В Норвегии осуществляется широкая программа мероприятий в связи с решением вопроса о размещении атомных электростанций под землей. Полученные на сегодняшний день данные свидетельствуют о технической возможности сооружения крупных промышленных атомных электростанций в подземных полостях. В результате применения такого метода строительства могут быть получены следующие преимущества: повысится безопасность и будут созданы более благоприятные условия для планирования городов, будет меньше ограничений для строительства с точки зрения окружающих условий, уменьшится влияние рельефа местности; повысится степень защиты предприятий с точки зрения гражданской обороны.

Экономические расчеты показывают, что единственные дополнительные расходы при сооружении атомных электростанций под землей по сравнению с обычным методом строитель-

ства вызваны стремлением получить дополнительные преимущества. В Норвегии накоплен значительный опыт решения различных проблем, который найдет применение при строительстве атомных электростанций под землей. Например, в 1959 году вступил в строй испытательный реактор HBWR в Халлене мощностью 25 МВт, на котором грунт используется в качестве основного защитного слоя. Большинство гидроэлектростанций в Норвегии также сооружены в подземных полостях. Проводимые в настоящее время научно-исследовательские и опытно-конструкторские работы имеют целью получение новых данных о способности различных видов породы удерживать продукты деления, определение проблем, связанных с радиоактивным загрязнением грунтовых вод, и установление ограничений в области гражданского строительства при создании больших полостей в различных геологических условиях. На основе результатов этих научно-исследовательских и опытно-конструкторских работ будет проведена дальнейшая техническая и экономическая оценка проблемы размещения атомных электростанций под землей, с тем чтобы получить всевозможные преимущества за счет повышения безопасности при использовании этого метода сооружения атомных электростанций.

EMPLAZAMIENTO SUBTERRANEO DE CENTRALES NUCLEARES EN NORUEGA.

Se está emprendiendo en Noruega un amplio programa de estudios en relación con el emplazamiento subterráneo de centrales nucleares. Los resultados obtenidos hasta el momento demuestran que es técnicamente posible emplazar grandes centrales nucleares comerciales en cuevas rocosas. Las ventajas que pueden obtenerse de este procedimiento son las siguientes: mejores condiciones de seguridad y posibilidad para emplazamientos urbanos; menos restricciones sobre los terrenos circundantes; influencia reducida sobre el paisaje; mejor protección desde el punto de vista de defensa civil. Estudios económicos demuestran que este procedimiento supone únicamente un incremento de costes marginal en relación con los métodos clásicos de emplazamiento.

En Noruega se ha adquirido una considerable experiencia sobre distintos problemas relacionados con la construcción subterránea de centrales nucleares. Así, el reactor de prueba de Halden, de 25 MW(t) y agua pesada en ebullición, se halla emplazado en una excavación rocosa que se utiliza como barrera de contención y ha estado en funcionamiento desde 1959. La mayoría de las nuevas centrales hidroeléctricas de Noruega también tienen este tipo de emplazamiento. Existe en la actualidad un programa de investigación y desarrollo con objeto de determinar la capacidad de las distintas clases de roca para retener los productos de fisión, identificar los problemas relacionados con la posible contaminación de aguas subterráneas, y estudiar los límites técnicos de construcción de grandes cavidades en diferentes estructuras de roca. Basándose en los resultados de este programa, se continuarán haciendo estimaciones técnicas y económicas de centrales subterráneas, con vistas a aprovechar al máximo la seguridad que ofrece este tipo de emplazamiento.

1. INTRODUCTION

In Norway, virtually all electric power is produced in hydroelectric stations. Most of the new stations are located underground, and they contribute 75% of the total power generation, which amounts to about 60 TWh/yr.¹

Extensive experience accumulated in the design and construction of underground plants — both power stations and industrial plants — has focused attention on the possibility of siting future nuclear power stations underground. As early as 1955, this contributed to the decision to build our 25-MW(th) test reactor, the Halden Reactor, underground.

The present paper reviews the work which the Institutt for Atomenergi (IFA), Norsk Hydro A. S. and the Norwegian Water Resources and Electricity Board (NVE) have performed together on the underground siting of nuclear power stations in Norway. Future research and development work in this field is briefly outlined, and relevant information on the Halden Reactor containment is presented.

It is believed that the work on the underground siting of nuclear power stations in Norway might also be of general interest to other countries.

¹ Terawatt-hour = 10⁹ kWh.

2. MOTIVATION

There are a number of incentives to study the underground siting of nuclear power stations in Norway.

A potential building site for such a station should have an optimal location with regard to power (and possibly heat) consumption, existing transmission grids, fresh-water and cooling-water resources, communications (harbours, railways, highways), etc. The topography and population pattern of the country is such that a number of areas that otherwise would be suitable for a nuclear station in the open are already densely populated or used (or to be used) for conventional industrial purposes. Few sites are available for which extensive landscaping and rock blasting would not be required.

For underground sites, inexpensive land with a low population density can be utilized, and the excavated rock may be used for preparing the necessary facilities above ground.

The geology at the potential building sites is, in general, well suited to underground siting. This, together with extensive experience and skill in rock excavation, results in relatively cheap constructions. The construction work will also be more independent of the weather conditions. Even if the construction cost of an underground plant is higher than for a corresponding plant above ground, this may be offset by other economic advantages, such as a cheaper lot and better location.

The underground siting of nuclear power stations is advantageous from the point of view of safety during major reactor accidents, including hypothetical accidents such as vessel rupture and core meltdown. There are also obvious advantages where the consequences of aircraft crashes and acts of war are concerned.

The underground siting of nuclear power stations can thus be beneficial from a purely economic point of view, as well with regard to safety and landscape preservation.

3. NECESSARY ASSESSMENTS

The need to assess the underground location of Norwegian nuclear power stations was recognized at an early stage, and two approaches were used.

The first was on a short-term basis and concerned the possible underground siting of our first nuclear power station. The containment systems evaluated are essentially the same as those for a plant erected above ground, and the rock serves only as a secondary containment. Such an underground plant therefore represents a minimum deviation from the standards applied to surface plants.

The second is a conservative approach, which stresses the need for assessment of an underground plant where optimal utilization of the containment properties of the rock is sought. Such a radical departure from common practice will necessarily require considerable research and development work and, of course, increased efforts in the engineering aspects and licensing of the first plants of this type.

The latter assessment will necessarily be longer term, and the work might not have too great a bearing on our first nuclear station.

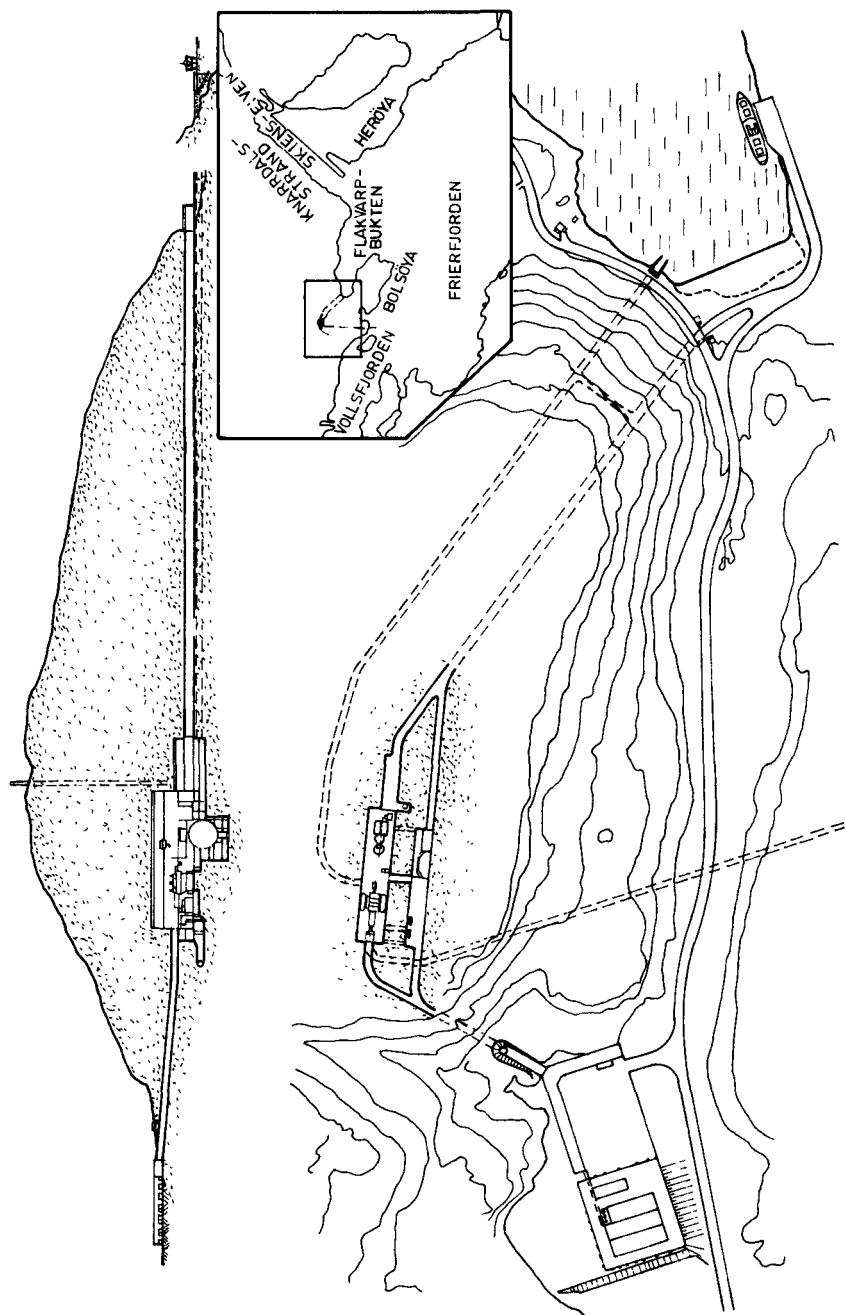


FIG. 1. 500-MW(e) BWR: layout.

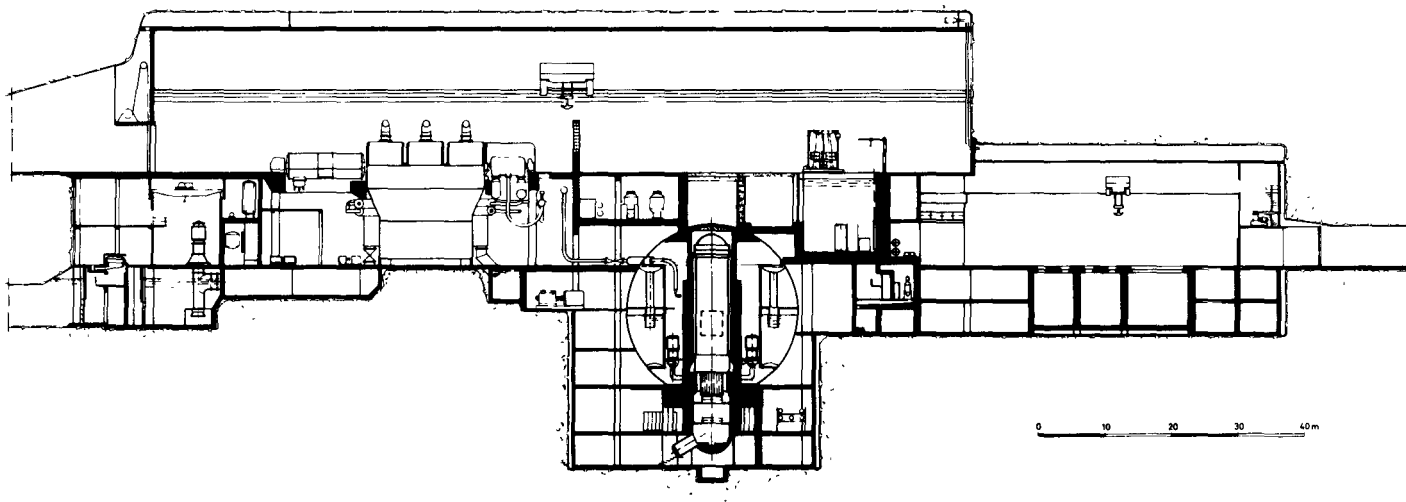


FIG. 2. 500-MW(e) BWR: reactor and turbine hall.

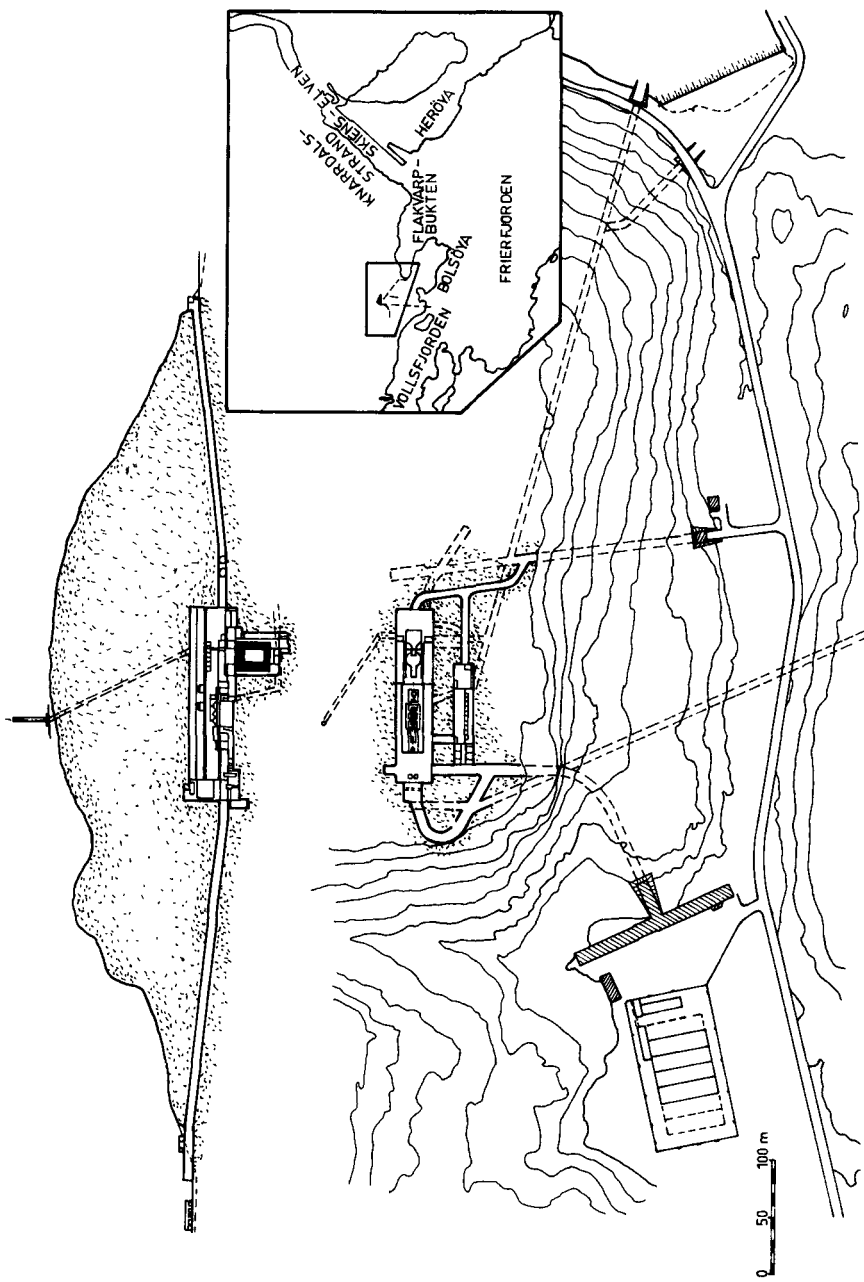


FIG. 3. 500-MW(e) AGR: layout.

4. SITING STUDIES PERFORMED

In the period 1966-1969, a co-operative power reactor study was carried out by Norsk Hydro, A.S., NVE and IFA [1]. Two reactor types were considered in this study: a boiling-water reactor (BWR) and an advanced gas-cooled reactor (AGR). Both reactors (500 MW(e)) were planned for two specific sites – a surface site and an underground site – and the construction costs were compared.

The site selected for the underground plant was in a dome-shaped rock formation south-west of Oslo. The drawings were based on information from suppliers of the reactor types.

4.1. The BWR station

The arrangement of the BWR station is shown in Figs 1 and 2. The reactor and turbogenerator are located in a hall with dimensions of 125 m × 25 m. The maximum height of the cave is 70 m. An adjacent hall (110 m × 15 m) contains the control room, transformers and electric auxiliaries. The office and workshop building is located above ground together with the switchyard.

Normal access to the power station is through the tunnel from the office and workshop building. Heavy equipment and fuel are transported through the tunnel emerging at the quay. The tunnels are roughly 200 m long.

The cooling-water intake is 30 m below sea level and the water is discharged over a spillway by the quay. Ventilation air and offgas pass through an inclined shaft from the station to a chimney on top of the hill.

The reactor, with its containment vessel and auxiliary systems, is located underneath a common main floor. There is easy access to the main floor over the reactor. An 8-m-high concrete wall acts as a biological shield for the radiation from the turbine hall.

The total volume of the power plant excavation is 265 000 m³.

4.2. The AGR station

The station layout is shown in Figs 3 and 4. The main entrance is through a 225-m tunnel to the turbine end of the main hall, and an auxiliary entrance is via a 250-m tunnel from the quay to the reactor end of the main hall. Branches of this tunnel accommodate the CO₂ storage plant and the solid-waste storage vault. The cooling-water inlet and outlet are basically the same as those described for the BWR plant.

The main hall (148 m × 30 m) has a maximum height of 83 m. To achieve a minimum span width in the hall, an AGR of the pod boiler type was chosen (the version of larger diameter – integral design reactor – was chosen for the surface site).

A separate hall (106 m × 16 m, maximum height 23 m) contains control room, electrical equipment, main transformers, etc. The office and auxiliary buildings, together with the switchyard, are located in the open adjacent to the entrance of the main access tunnel. The total volume of the plant excavation is 275 000 m³.

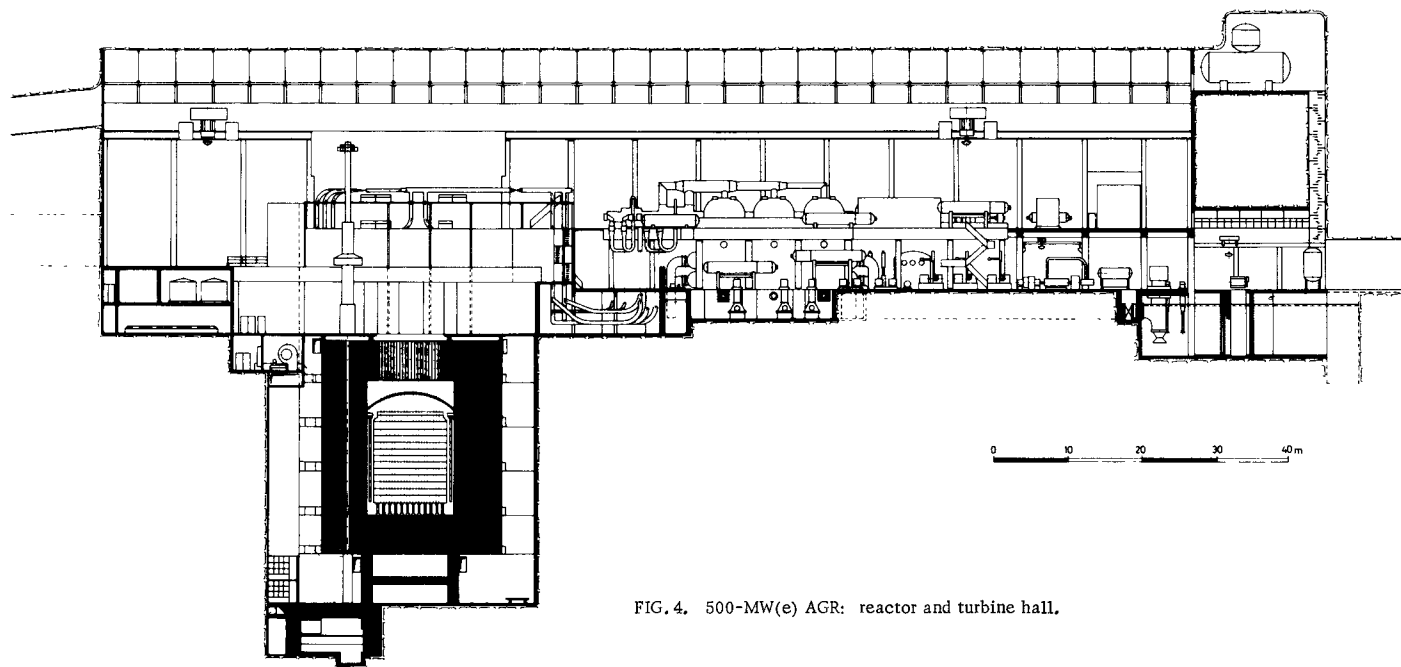


FIG. 4. 500-MW(e) AGR: reactor and turbine hall.

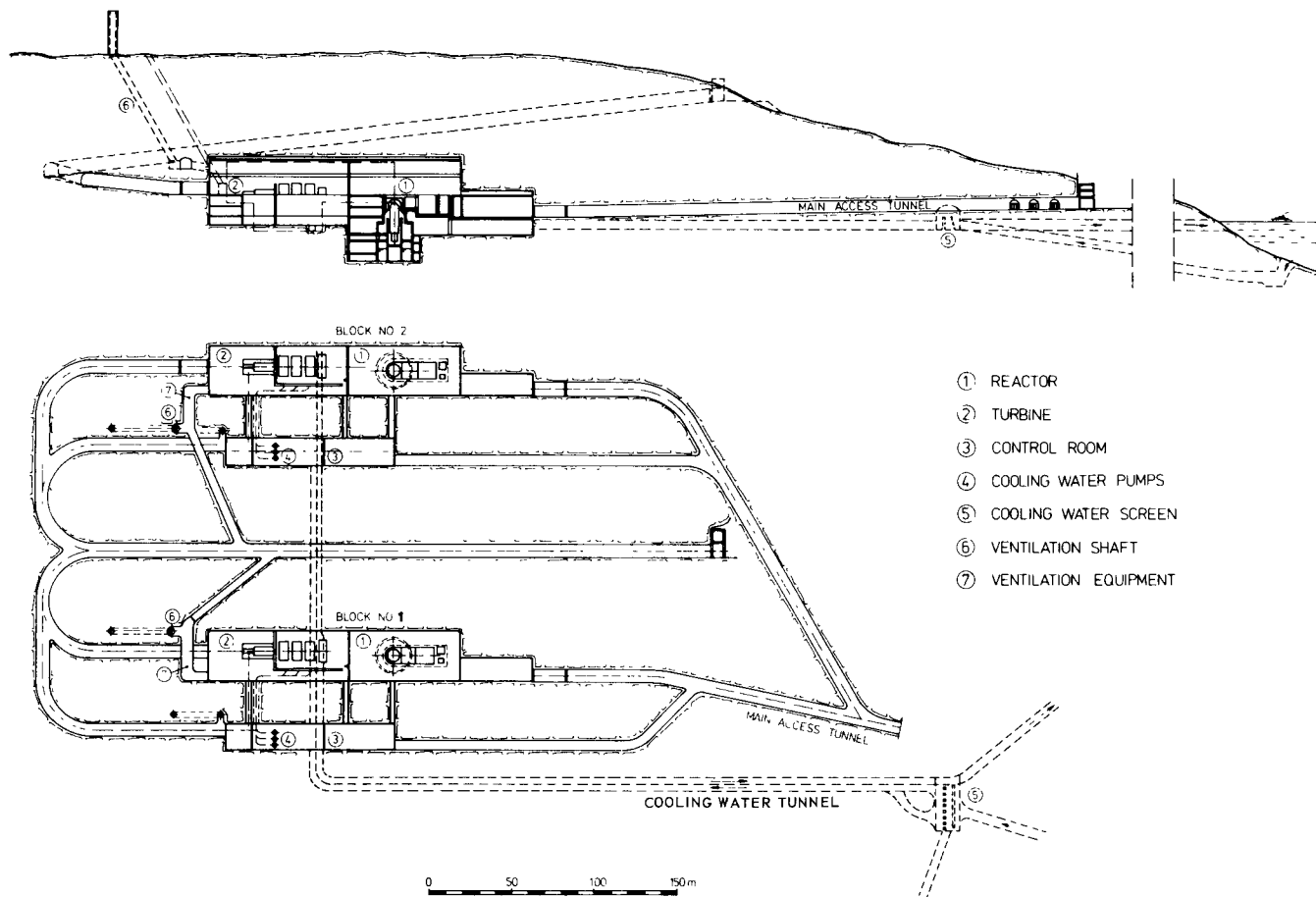


FIG. 5. Twin 500-MW(e) BWR station: layout.

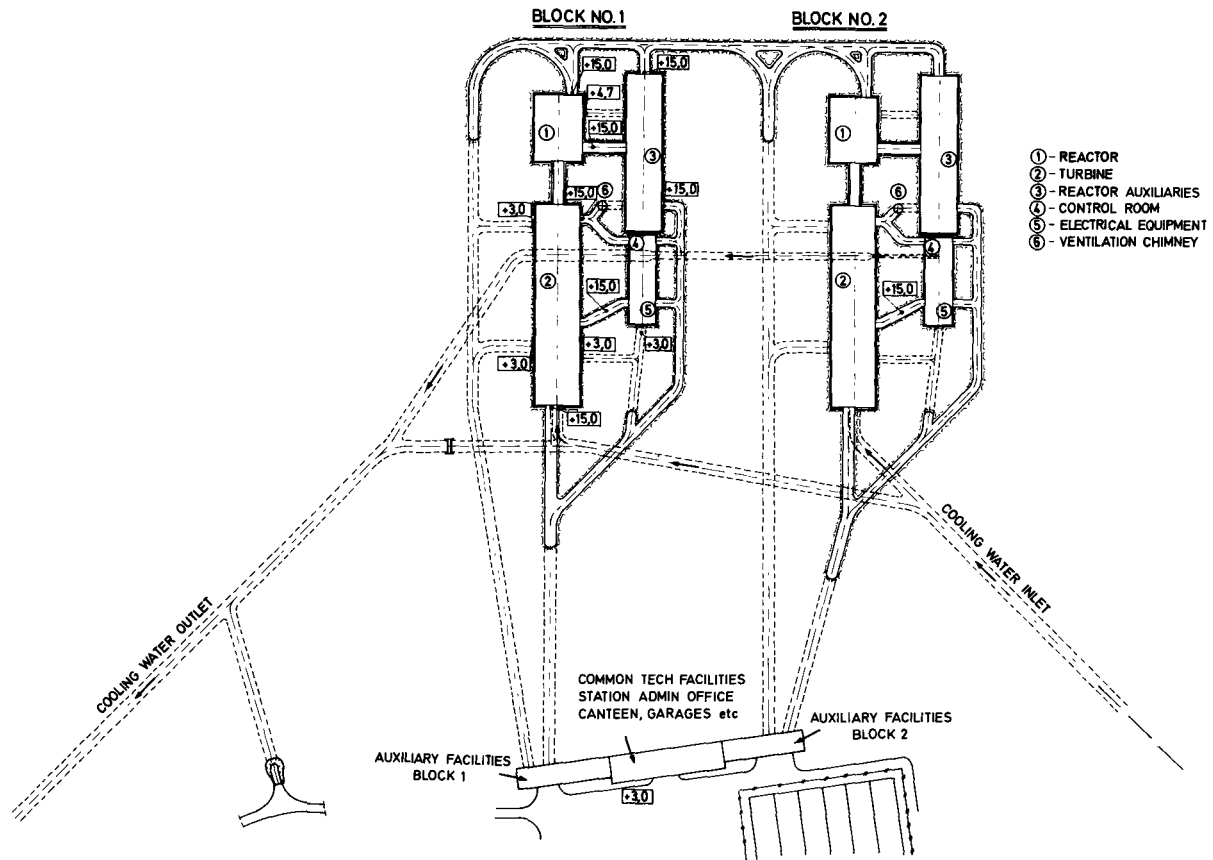


FIG. 6. Twin 500-MW(e) PWR station: layout.

4.3. Discussion

The location and orientation of the halls are strongly influenced by the following considerations: geotechnical properties of the rock; rock overburden; arrangement of cooling-water tunnels to avoid recirculation of the heated water, easy access and transport during excavation, construction and operation (handling excavated masses with cranes should be avoided to the greatest possible extent). These requirements result in a quite extensive tunnel system, which can be seen from the layout drawings.

These studies indicated that underground siting increased the construction time by 10-12 months, and the increase in the total construction costs, including interest, was approximately 4% as compared with a corresponding surface plant.

In this connection it should be pointed out that the plants above ground were designed without special civil defence measures, whereas the underground alternatives meet the most stringent requirements.

The overall economics of this type of underground station will, of course, vary from site to site, and it cannot be excluded that such plants could be competitive with a standard surface plant with all factors taken into consideration.

After completion of the reference study, the preplanning for our first nuclear power station continued. Some potential sites in the eastern part of Norway were surveyed, including some underground sites. Figure 5 shows the arrangement of a twin 500-MW(e) BWR plant at one of the potential sites and Fig. 6 shows the same arrangement of a PWR plant.

A geological survey program at one of the underground sites is scheduled for the summer of 1971 to establish the rock properties at the site from the point of view of construction.

Research has been started on the acceptable span widths for underground halls. The seismic problems connected with the excavation of a second reactor hall at the site of an already operating reactor will also be investigated.

5. THE USE OF UNCLAD ROCK AS CONTAINMENT

The approach to underground siting described above is conservative, and it is felt that a more radical approach might provide interesting results. Experience with the Halden reactor may have some relevance in this context.

5.1. The rock containment for the Halden reactor

The 25-MW(th) Halden reactor was built by the Institutt for Atomenergi in the period 1955-1958. Since 1958 it has been operated as a test station under the auspices of ENEA [2]. The reactor is located in a cavern in a hillside outside Halden in the south-east of Norway. Figure 7 shows the layout of the reactor.

The rock in which the reactor hall is situated consists of gneiss. Fissures formed by dislocations are distributed through the rock and

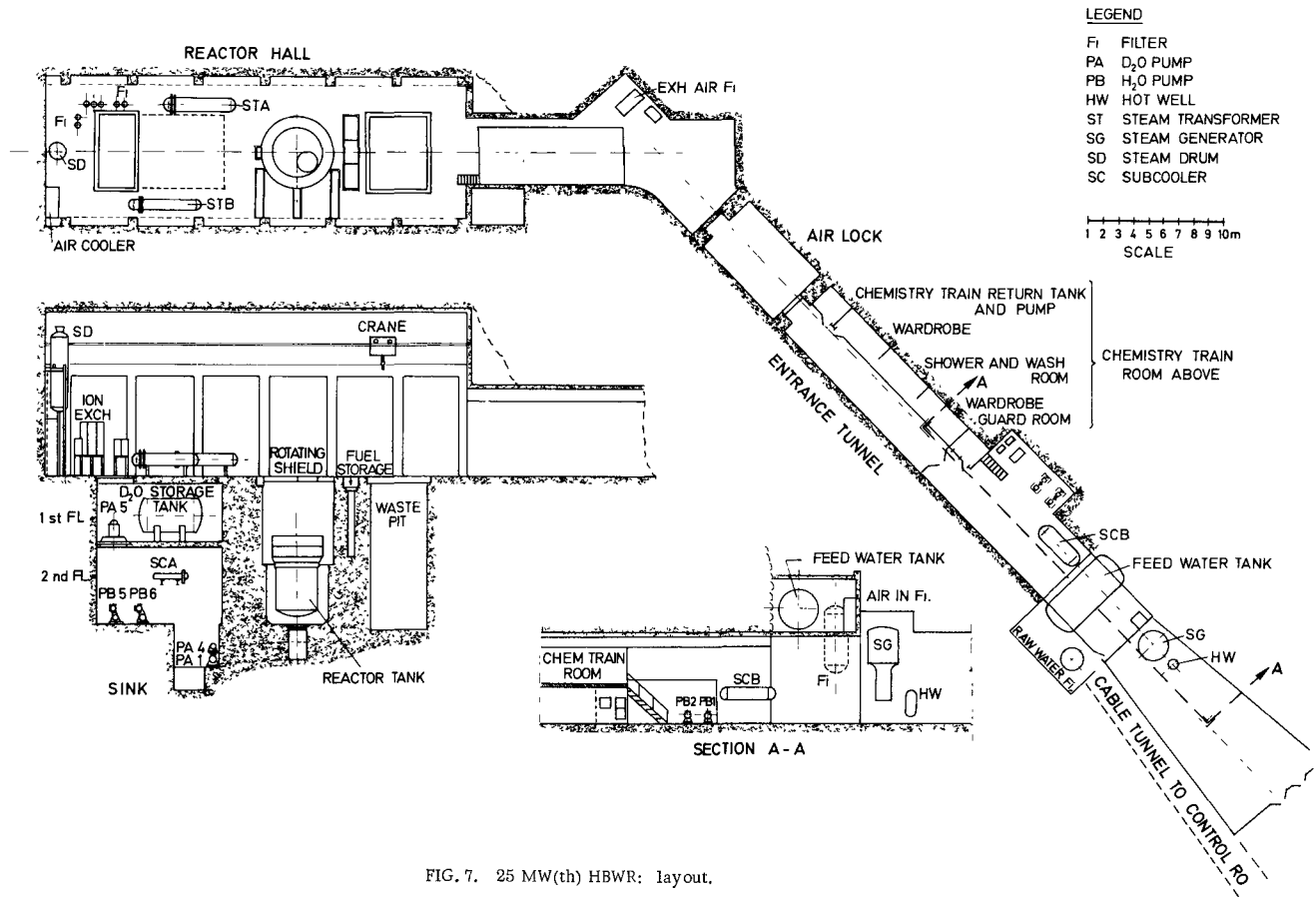


FIG. 7. 25 MW(th) HBWR: layout.

are filled with stone powder and chloritic materials formed by the leaching of the gneiss. From 5 to 10% of the total material in the cracks is montmorillonite clay.

The reactor area is entirely underlaid with bedrock, which permits very little sub-surface drainage. Some drainage does occur, however, through cracks and fissures in the gneiss. These fissures are parallel to the dip of the rock and lead downwards towards the sea. Ground water flows slowly into the reactor hall and is collected in a sink at the lowest part of the excavation, 1.2 m above sea level, at a rate of 25 m³/d. This inflow is quite independent of the weather conditions.

The cavern containing the primary system has a volume of 4600 m³. It is lined with concrete and the walls are painted. There are open cavities between the concrete walls and the rock at several places, in particular above the vault in the hall, where the cavities communicate with the hall through a port. The minimum rock overburden (in the air-lock area) is 30 m.

In the vicinity of the reactor hall and the access tunnel, the rock is cracked due to the blasting, which results in relatively good communication between the various cavities behind the walls. The tunnel entrance is closed by two 1-m-thick, reinforced-concrete walls. These walls are covered by an epoxy lining, and the cracked rock around the airlock is injected with clay and concrete through a series of holes up to 5 m deep. This work took place after completion of the construction. By proper pre-planning and design of the airlock for injection, better leaktightness could have been achieved.

There is a manually operated emergency spray system for the hall, and an automatically started spray system in a pit, through which air, in the event of an accident, is circulated by fans. The fans return the air to the hall, one of them through an impregnated charcoal filter.

Most of the leakage from the plant ends up in the tunnels which are provided with two filter systems. One of these filters air from the vault above the tunnel and the other air from the tunnel itself, which is closed by a wooden structure at the exit. Impregnated charcoal is used as a filter.

An emergency purification system is available to take care of the spray water and the continuous inleakage of water through the rock in the event of an accident with activity release to the reactor hall.

The gas leakage rate from the cavern has been measured on several occasions and amounts to about 20 m³/h per metre w.g. overpressure, and shows little variation with time. About 25% of the outleakage passes the airlock room, mainly through its uncoated side walls. The main leakage from the hall ends up in the cracked zone and the vault above the tunnel outside the airlock.

The primary system contains 14 tons of heavy water, most of which is at operating temperature (240°C).

In the reference accident, the coolant is lost through rupture of a pipe at the bottom of the reactor vessel, whereby a peak pressure of 1.3 atm. is reached in the reactor hall. The subsequent temperature rise in the small, open latticed core is lessened by the great heat losses.

It is assumed that the activity release from the core takes place in two batches – a small one immediately after the rupture of the pipe, and a major one which takes place after a certain delay. It is further

assumed that the iodine content in the reactor hall decreases at the same rate as the steam content in the hall atmosphere during the condensation. The methyl fraction, representing 5% of the iodine inventory is, however, only considered to be removed in the recirculation air filter for the spray pond.

It is assumed that the cavities and cracks behind the concrete lining are initially filled with air of the same concentration as that in the reactor hall immediately after the loss of coolant, and that back expansion of this air to the hall during the pressure decay in the latter keeps air with higher activity concentrations away from the cavities during the initial pressure decrease.

The gas leakage is taken as proportional to the overpressure in the hall. It is assumed that half of the outleakage passes from the cavities behind the walls of the reactor hall and to the surface of the hill, whereby 80% of the iodine activity is deposited in the rock. The delay effects during this passage are neglected.

The remaining outleakage is considered to end up in the tunnel area, where 95% of the iodine is filtered out before discharge to the stack.

The possibility of releasing air from the reactor hall during the initial phase of the accident has been studied. The reduced air content would facilitate quicker pressure reduction and a shorter overpressure period in the hall. This is of particular importance for a system where there are significant time delays in the activity release and containment leakage. One advantage of such a system is that malfunctioning of the containment penetration equipment will not be so critical.

For the Halden reactor, a pressure release system was designed whereby air would be released through a water lute into a pond after an accident. The system was, however, never made operational, partly because of difficulties in guaranteeing the washout effects during the release.

The leakage properties of the rock in Halden are rather special and not representative of the rock at our potential sites for power reactors. By application of modern techniques, a more satisfactory airlock could also have been constructed. The work with the Halden reactor containment has, however, contributed to our knowledge of rock containment and indicated that the use of the rock itself as a leakage barrier may be of interest for our future power reactors.

5.2. Future research program

Early this year, IFA initiated a study, in co-operation with NVE, on the use of unclad rock as containment for light-water reactors. This study aims at determining: (a) leaktightness requirements for the reactor hall and the prospects of fulfilling these requirements; (b) the risk of groundwater contamination; (c) the safety advantages as compared with a more conservative plant; and (d) the economics as compared with a more conservative plant.

When completed, the rock cavern for a reactor will have to pass a leakage test to prove that a predetermined leak rate is not exceeded. The permissible leak rate will have to be settled in advance and will depend on the distribution of leakage to the rock surface and the tunnel system. One way to handle this problem is to make the calculations as

if all leakage went to the rock surface or, alternatively, to the tunnel system. The leak path that shows the most serious consequences for the environment will determine the leaktightness requirements for the hall.

For our sites, the minimum rock overburden inside the airlock will be of the order of 40-80. The leak path from the hall to the surface will be of at least the same magnitude. This means low leak rates, long time delays during passage through the rock, and strong filtering effects. Groundwater in the rock will limit the gas leakage. The study aims at clearing up these problems by means of a series of field experiments at potential sites. These experiments will be co-ordinated with the geotechnical surveys.

Leakage through the airlocks ends up in the tunnel system, where it can be filtered and/or temporarily stored for release through the stack, as weather conditions permit. It can be seen from Figs 5 and 6 that the possibility of storing active gases in the tunnel system and rooms outside the airlocks should be quite promising.

The integrated leakage is dependent on the magnitude and duration of the overpressure in the hall after an accident. In a rock containment of this type, the volume and wall surfaces in the containment will be relatively great, and this results in a low peak pressure and significant natural cooling effects after an accident. This will be studied together with the possibility of a controlled bleed-off from the hall to bring the pressure more quickly down to atmospheric.

The above investigations will help to determine the leaktightness requirements of a reactor hall in rock. Methods and means of determining whether a given site is suitable for such an underground containment will be looked into. The possibility of getting a reactor hall that does not satisfy the leaktightness requirements into a satisfactory condition will also be considered. The risk of groundwater contamination and the means of treating contaminated drainage water after a loss-of-coolant accident will be evaluated.

An important aspect of the study is to investigate the differences in safety characteristics between stations above ground and stations underground with and without rock as primary containment. The risk in connection with malfunctioning of emergency equipment, including core meltdown and vessel rupture, will also be compared for the above containment systems.

Finally, the relative economics for the various alternatives will be studied.

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- [1] NORSK HYDRO, NVE-STATSKRAFTVERKENE, Institutt for Atomenergi, Nuclear Power Plant in Norway, September 1969.
- [2] JAMNE, E., SIVERTS, J., Description of the HBWR Plant, HPR 95, December 1967.

DISCUSSION ON AGENDA ITEM 1.5

Review of safety aspects of nuclear power plants

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/173 Italy Presented by R. Lepore

P/272 Belgium Presented by H. Dresse and P. Dozinél

P. COSTE: In the Tihange station referred to by Mr. Dresse what is the static strength of the biological shielding around the primary circuit? Is the design of the concrete reinforcement affected by the strength requirements?

H. DRESSE: The shielding is designed to withstand the transient pressure arising in the event of an accident. The pressure arising in the cylindrical part around the tank is limited to about 5 kg/cm² and beneath the tank it is limited to half this value. The concrete reinforcement is designed for the purpose and, in particular, the density of the braces is affected.

N. J. PALLADINO: I should like to ask Mr. Lepore whether the liquid-poison sparger at the Garigliano plant had any core-spray function during emergency core-cooling?

R. LEPORE: The liquid-poison sparger had no connection with emergency core-cooling. The only purpose of the liquid-poison system was to bring the reactor slowly to cold sub-critical, starting from hot sub-critical, if the control rods failed to scram.

D. OKRENT: On what basis was it decided that an earthquake scram was needed? What is the design basis for initiating scram and why was the particular scram setpoint selected?

R. LEPORE: I shall ask Mr. Iansiti, the co-author of Paper P/173, to answer this question.

E. IANSITI: The resistance of the plant to earthquakes has been evaluated, but a more complete evaluation is at present being undertaken. In the meantime, the seismic scram was requested. Using a computer program with seismic acceleration data as input we estimated that, with a low seismic-scram setting, the control rods would be partially in the core when large displacement occurred. A report on this study is available.

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/039 USA Presented by G. F. Brockett

P/365 FRG Presented by K. Traube

G. HAKE: Mr. Brockett states in Paper P/039 that hypothetical events are used to evaluate design adequacy. Could he please elaborate on this statement.

G. F. BROCKETT: The events postulated vary from one design to another. The assumption of an arbitrary ramp rate of 50 cents per second, for example, implies energy releases that are large enough to provide an adequate test of containment. Another example is a loss of peaking power followed by a failure to scram; it should be possible to design against this accident, which should therefore be considered hypothetical.

F. C. BOYD: Could Mr. Traube say what is the attitude in Germany concerning the need to design containments which will cope with failure of the reactor pressure vessel?

K. TRAUBE: In connection with the safety reviews for two new plants, a sub-committee of the German Safety Commission decided about a year ago that rupture of the pressure vessel need not be considered beyond the fracture of a large pipe. In the case of the plant I mentioned it was decided that more research work pertaining to urban siting was necessary before licensing.

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/041 USA Presented by A.J. Pressesky

P/027 Netherlands

There was no discussion on these papers.

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/038 USA Presented by C.K. Beck

P/364 FRG Presented by O. Kellermann

P/304 Sweden Presented by G. Schultz

P/474 UK Presented by J.R. Harrison

P/475 UK

F. C. BOYD: My question is to Mr. Kellermann (P/364). In your paper you refer to a maximum level of 2 rem/30 years for the population dose due to nuclear power. In establishing the operating limits for individual plants, do you consider a hypothetical individual, on or near the boundary, an average population, or a critical group?

O. KELLERMANN: We consider a hypothetical individual at the boundary of the plant site and we take into account all plants discharging radioactive effluents in the neighbourhood. If, say, it is proposed to construct four plants at a chosen site, the permitted effluent per plant is a quarter of what it would be for one single plant.

F. C. BOYD: To ensure that the level is not exceeded, do you also require analysis of data on actual releases in combination with information on meteorological conditions, water usage, food chain, etc?

O. KELLERMANN: Meteorological conditions are taken into account in evaluating the operational and accidental releases of radioactive products. Effects due to the ecological chain and direct irradiation through

inhalation and ingestion are taken into account in the codes that have been developed. Actual releases have been measured and compared with the calculated values.

J. FURET: I should like to ask Mr. Kellermann whether the criterion of separation of the operation monitoring instrumentation from the safety control instrumentation also applies to light-water stations of the boiling-water type?

O. KELLERMANN: The criteria are valid for pressurized- as well as for boiling-water reactors. They are based on an IEC (International Electrical Commission) recommendation.

J. FURET: Does the criterion also extend to the detectors in the core?

O. KELLERMANN: The criterion of separation of process and safety instrumentation is not applicable to reactors in operation or under construction. However, the trend is to have separate lines and detectors for certain essential safety functions. We will keep in close contact with the relevant international bodies in this connection.

F. C. BOYD: I should like to ask Mr. Beck (P/038) to what extent the new requirement to perform a risk/benefit evaluation will cause the USAEC licensing regulatory group to turn to a probability approach of reactor safety evaluation.

C. K. BECK: We have just established the regulatory framework of the procedures we will follow in response to these new responsibilities which have been thrust upon us, and it is quite clear that we will have to use whatever quantitative means are available. In the process of evaluating accidents, for example, there is no doubt that the probability approach will be used and perhaps much more extensively than it is used at present in our safety analyses. Just how far we shall go or how well we can do it remains to be seen.

C. SENNIS: This is another question to Mr. Beck. In some instances the USAEC general design criteria make explicit reference to the single failure criterion. At the same time, it is our understanding that the USAEC and the Advisory Committee (ACRS) have requested applicants to give consideration to the course of events in cases of abnormal operation, taking into account multiple system failures. To what extent is this multiple failure analysis featured in present safety reviews? Do you foresee a possible future definition of safety criteria on the basis of multiple system failure, possibly related to reliability considerations?

C. K. BECK: There are instances in the prescribed procedures for analysing and evaluating the hypothetical situations relating to design basis accidents. In these prescribed formulations, we not only specify the initiating failure of an incident but also postulate the possibility of failure of back-up systems that might be called on to help to terminate the accident. I do not know to what extent the quantitative description of probability will evolve in these analyses. Certainly there is one feature of the situation that may change as we proceed, and that is that in our safety analyses we have deliberately made rather pessimistic, conservative assumptions in order to be sure that we are well on the safe side as regards protection against these serious accidents. The more realistic appraisal of probabilities in environmental effects may constitute a different approach to the problem, where of necessity we must try to be as realistic as possible. Two parallel considerations might very well evolve, each serving a different purpose and each justified in its own right.

N. AYBERS: I, too, would like to put a question to Mr. Beck. What significance should be attached to the probability of component failure in safety analysis in view of the fact that some components have not yet been long in use?

C.K. BECK: We have not yet tried to quantify the failure of components in our safety analyses. We believe that probability analyses serve very useful purposes in comparisons between systems, but we have not yet found it a profitable procedure to try to put numbers on the failure probabilities of these components which, as you point out, have only a short history. We are at present working out a systematic plan for collecting information on system failures and perhaps as we gain more experience probability will play a bigger part.

C. O'FARRELLY: I would like to ask Mr. Beck a fundamental question. What potential risks does he consider technologists are entitled to take? For a poison, the best measure of risk is probably its toxic potential, defined as the quantity present divided by the lethal dose. Let me take a practical example. There are chlorine tanks in the United Kingdom with a capacity of 5000 tons. With a lethal dose of 1 gram this represents 5000 million lethal doses. Failure of that tank is not going to kill 5000 million people but, depending on the weather and population conditions, it is going to kill a large number. Within a radius of 50 kilometres there are two cities, each with a population of about 3 million people. A nuclear reactor will have a toxic potential 1 to 2 orders of magnitude higher than this. What risks are we entitled to take, especially if they are not absolutely essential and can be diminished?

C.K. BECK: I have no quantitative answer to this. The question as to how far we should go has to be tackled in a subjective or intuitive way until more quantitative approaches become available. In the meantime we are trying to determine the capabilities of our technology. We are still at the beginning, even though some reactors have been in operation for a while. We still have a very new technology and our objective has been to ensure as far as possible that we are on the safe side.

A.K. GANGULY: I have two questions for Mr. Beck. Could he please elaborate a little on the reasons for setting the dose of 5 millirem in the public domain as the acceptable limit for an operational BWR station? My second question is: what are the primary criteria in the United States of America for licensing multiple-reactor stations, where various facilities are shared by the reactors?

C.K. BECK: The dose limit of 5 millirem was fixed in accordance with the ICRP recommendation that radiation exposure should be kept as low as practicable. In reply to your second question, I can only say that we have a basic rule that when more than one station is built at a site, they should be built in such a way that an accident in one would not give rise to an accident in the other. Where they share facilities, it is necessary to ensure that failure of one system could not jeopardize both together. There are some cases in which these criteria can be met with shared facilities.

A.K. GANGULY: Are there any stipulations regarding the acceptable dose in the public domain from fuel-reprocessing plants?

C.K. BECK: We have no specifications for fuel-reprocessing plants. We would follow the same general principles in approaching that problem.

V.A. SARABHAI: I should like to ask Mr. Beck how he takes account of the man/machine interaction in this safety analysis? A case in point arises in the inerting of the containment of BLW reactors. Once it is inert, the man/reactor relationship becomes quite different.

C.K. BECK: We do have some guidelines on this and as regards inerting the containment we are balancing the possibility of some reduction in safety on the one hand with what appears to be a much larger gain in safety on the other. If we do not inert the containment, certain serious situations could arise. On the other hand, by inerting the containment we facilitate and improve the inspections of the equipment inside the containment. I think it is a matter of judgement. There are other ways in which we try to take account of this man/machine interaction, for instance by the use of automatic systems. We have not quantified this too far, but we try to take it into account in a qualitative way.

S.O.W. BERGSTRÖM: Could Mr. Beck comment on the fact that the USAEC decided to use the dose rate at the boundary as the main criterion instead of a man-rem dose? We have tried the 5 millirem per year dose limit on different population densities and found that the man-rem dose could vary from 100 man-rem per year to 10 000 man-rem per year. It therefore seems that the 400 man-rem value given in your guidelines is not always in line with the 5 millirem per year.

C.K. BECK: There are two different problems involved in your question. One is the choice of a site. The man-rem is certainly a very important element in deciding whether a reactor should be placed at a particular site, in relation to its effluent possibilities. I think we chose the millirem concept simply on the basis of the effluents which were possible in the case of these types of reactors.

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

<i>P/579</i>	<i>France</i>	<i>Presented by J. Bourgeois</i>
<i>P/150</i>	<i>Canada</i>	
<i>P/477</i>	<i>UK</i>	<i>Presented by F.R. Farmer</i>
<i>P/040</i>	<i>USA</i>	<i>Presented by P.A. Morris</i>
<i>P/734</i>	<i>Spain</i>	<i>Presented by A. Alonso Santos</i>
<i>P/476</i>	<i>UK</i>	

R.H. CAMPBELL: It appears to some of us that the safety considerations for water reactors and fast reactors are not being dealt with in an identical manner. Mr. Pressesky (P/041), for instance, talked of a hypothetical accident in a fast reactor, postulating tripping of the pumps with failure of the reactor to scram. Also, in fast reactor safety, very great emphasis is placed on SPERT (Special Power Excursion Reactor Tests) type accidents. I should therefore like to ask the following general question. Does the difference arise because of the very good intrinsic safety features of the sodium-cooled reactor, e.g. the low pressure and the existence of a large heat sink from which it is not necessary to remove the decay heat for a long period (so that the reactor safety is not dependent on the ability to start plant up quickly)?

F. R. FARMER: I believe that safety criteria have grown up historically and we have heard from speakers of the United States of America that their criteria are up-dated from time to time, but they still emerge from a previously set pattern. We had foreseen some difficulty in introducing new reactors, and in 1965-67 we developed the view that all accidents were possible, some more than others, and that they should be considered in relation to their likelihood. It is my belief that new reactor systems will have to show higher safety capability than existing systems, and I further believe that some designs of fast reactors will do this.

J. BOURGEOIS: In reply to Mr. Campbell's question I should like to say that the French method of analysis is not related to any specific, basic accident; the safety evaluation is the same for water reactors and fast reactors. In both cases the evaluation is carried out using the same method: continuous examination of the availability of barriers. Special assumptions can be considered for the different types after each analysis with a view to estimating the behaviour of fission products in the case of a very serious accident.

F. VELONA: Can Mr. Morris (P/040) say for how long the 'interim criteria' he mentioned are going to be maintained as such? Are the codes for the calculations relating to the emergency core-cooling systems also 'interim'?

P. A. MORRIS: Significant new information, either experimental or theoretical, will have to be obtained before any important changes in the criteria can be made. This does not mean, however, that all the research and development work will have to be completed before changes can be made. As results are obtained during the next few years, changes will be considered. This applies both to criteria and codes.

G. HAKE: The Canadian Paper P/150 describes a reactor safety assessment policy which is applicable regardless of reactor type, whilst the United States, German and United Kingdom reports related only to certain types. In the case of the United States of America does this mean that in the matter of quality assurance, periodic inspection, etc., different philosophies will be adopted as new types are introduced?

P. A. MORRIS: No. The United States Atomic Energy Commission makes a conscious effort to ensure a consistent approach to all reactor types. For instance, there is a program for checking for common mode failures and for the consequences of anticipated transients or faults whose probability is significantly higher than that of the 'maximum credible accident'.

J. EIBENSCHUTZ: What differences exist in the analytical codes for emergency core cooling, as between the four North American manufacturers? There seems to be a difference of standard which could give rise to problems if the reactor purchaser invites bids from manufacturers without adequate experience.

P. A. MORRIS: The AEC has approved an evaluation model for General Electric boiling-water reactors, based on General Electric computer codes. For PWR reactors the AEC has approved an evaluation model using Westinghouse codes and other codes developed at the reactor testing station in Idaho. Computer codes have been used by Combustion Engineering and Babcock & Wilcox for their designs, but the AEC has not yet approved evaluation models for these computer codes. The companies are free to use the AEC codes if they wish. All manufacturers are absolutely convinced that their emergency core-cooling systems are adequate.

S. O. W. BERGSTRÖM: Would anyone like to comment on the need for further experimental clarification of the physics and chemistry of molten cores?

P. A. MORRIS: The AEC regulatory position is still quite close to the recommendations of the Ergen Report published in 1967. Priority should be assigned first of all to accident prevention, then to assurance of an adequate emergency core-cooling system, with studies of lower priority to investigate the possibilities of designs which could accommodate a molten core.

O. KELLERMANN: In answer to Mr. Bergström's question, I should like to say that in Germany a working group representing the Minister of Education and Science is conducting studies on core melt-down, which may lead to various experiments. The results of such experiments could help in the design of engineered safeguards for fast breeder reactors.

J. ESTAPÉ ARNAU: I have a question to Mr. Alonso Santos (P/734). Many countries will be experiencing considerable difficulties in the near future in the siting of nuclear power stations. This applies especially to countries with ambitious nuclear programs where there are no big rivers or where the coastal areas are populated by tourists. In view of this do you not think it necessary to extend the right of expropriation in order to set up some administrative mechanism for providing sites for future power stations?

A. ALONSO SANTOS: In areas where there are no big rivers and where the coasts are temporarily or permanently densely populated the siting of nuclear power stations will be a serious problem; it is thus understandable that the electric power companies should want to have rights very soon to the sites they require.

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/305 Sweden Presented by J. Rydberg

P/576 France Presented by L. Grall

P/548 CSSR

P/225 Japan Presented by Y. Ando

A. J. PRESSESKY: I should like to ask Mr. Rydberg (P/305) whether anybody in Sweden has evaluated and published results on molten-core material/water steam explosions for the design in which a pool of water is provided in the bottom of the containment?

J. RYDBERG: To my knowledge, no such studies have been published, but there is a Government-sponsored committee working on urban site criteria and it may also evaluate the molten-core concept.

F. C. BOYD: You apparently use United States safety criteria for the Marviken heavy-water reactor. I do not understand this because the USAEC acknowledges that its criteria have been developed for light-water reactors. Was there no attempt to develop your own safety criteria in general and for the Marviken HWR in particular?

J. RYDBERG: Perhaps Mr. Carlbon will answer this question.

L. CARLBOM: The real reason was that there was no interest on the part of the utilities and the reactor industry in heavy-water reactors when the reactor was nearing completion.

A.K. GANGULY: In the development of containment philosophies and measures against design basis accident releases, there seems to be a tendency to consider only the release of airborne activities. What about the large volumes of contaminated liquids involved in some of these accidents and in post-accident operations? What stipulations exist regarding water leakage or seepage rates.

J. RYDBERG: The permitted release limits for liquid wastes are set by the authorities, i. e. by the Water Court.

B. COX: Mr. Grall (P/576) implied that stress corrosion cracking in austenitic alloys represents a safety problem rather than a maintenance problem. My understanding of the situation is that stress corrosion cracking (SCC) is likely to lead only to leaking tubes, which can be plugged or replaced, and it is therefore a maintenance problem; it should not lead to any catastrophic failure of the tube, which might make it a safety problem. Have you any evidence to support your contention that SCC is a safety problem?

L. GRALL: Stress corrosion cracking may lead to extensive fracturing and not just to local leakage, so that a safety problem may be involved.

D. OKRENT: In connection with Mr. Cox's question, I would just like to observe that for analysis of the behaviour of emergency core-cooling systems during postulated loss-of-coolant-accidents on PWR's, it is of interest to know the extent of leakage between the primary and secondary systems in view of its possible effects on reflooding.

B. SAITCEVSKY: I should like to ask Mr. Ando (P/225) whether the Japanese program on light-water reactor safety has led to a change in attitude regarding the maximum credible accident, i. e. the rupture of a pipe in the primary circuit?

Y. ANDO: We have been considering the maximum credible accident in two stages in Japan: (1) serious, technically possible accidents which would have the maximum radiological effect on the environment; and (2) hypothetical accidents which are not possible and are considered only for hazards analysis purposes; here one of the post-accident back-up systems is assumed to be ineffective.

This approach will not be changed for some time. For light-water reactors, the accidents corresponding to the above criteria are:

	Serious accident	Hypothetical accident
BWR	(1) Main steam pipe rupture (2) Loss of coolant (3) Rupture of off-gas storage tank	Ditto, neglecting the effect of emergency core-cooling, to consider 100% fuel melt
PWR	(1) Loss of coolant (2) Piping rupture in the steam generator	Ditto, neglecting the effect of emergency core-cooling, to consider 100% fuel melt

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/226 Japan Presented by H. Shibata

P/672 Switzerland Presented by W. Novak

M. R. SRINIVASAN: I would like to ask the authors of Paper P/226 whether they could indicate the additional cost of making the design aseismic at different values of horizontal ground acceleration, for a reference site and a particular type of reactor.

H. SHIBATA: It is very difficult to say, because we have not compared the cost of an earthquake-resistant nuclear power plant with the cost of the same type of plant in a non-seismic zone. I think the additional cost should be between 5% and 15%. More important than the cost increase is the fact that there may exist an upper limit of ground acceleration for design purposes.

N. J. PALLADINO: Is there any obligation in Japan to install strong motion accelerographs at nuclear power plants to evaluate the effect of possible earthquakes at or near the plants? If so, how many are required? If not, how are the effects evaluated?

H. SHIBATA: There is no official requirement to install strong motion accelerographs at nuclear power plants in Japan, but it is recommended. Some plants have installed three of them, and one plant has more than twenty instruments for this purpose. The method of evaluating the effects on the various plant components is under discussion.

A. K. GANGULY: Do you permit the construction of nuclear power plants on an alluvial foundation in zones of mild seismic activity in Japan?

H. SHIBATA: At present, the construction of nuclear power plants on alluvial soil is prohibited. The Tokai Power Plant of the Japan Atomic Power Company, a 166 MW(e) gas-cooled power reactor, was built on caissons on sandy mud stone.

A. K. GANGULY: Perhaps Mr. Morris could comment on the practice in the United States of America.

P. A. MORRIS: In the United States we do permit the location of nuclear plants on alluvia, as opposed to bed rock. However, the foundation soil may have to be specially prepared, for example, by compaction or by the use of piles.

A. J. PRESSESKEY: I should like to ask Mr. Novak (P/672) whether the rocking support concept is considered applicable for higher acceleration values such as $\frac{1}{2}$ or $\frac{3}{4}$ g. What are the limits with respect to overturning of the rolling supports?

W. NOVAK: If φ is the angle of tilt, the system will roll for $(dy/dx) \varphi < \mu$ (~ 0.1) and will slip for $(dy/dx) \varphi \geq \mu$. The limiting value of φ is 5.7° .

H. SHIBATA: In this connection perhaps I might point out that the characteristics of strong earthquakes depend greatly on the site conditions. The vertical motion, for example, exceeds the horizontal motion, and carries vibrations whose frequency may be over 70 Hz with an amplitude of 30-50 cm.

P. VYCHYTIL: I should like to ask both Mr. Novak and Mr. Shibata what safety criteria they have for the vertical acceleration component?

H. SHIBATA: As I mentioned in my presentation, for the design of class-A items one half of the horizontal motion should be taken. Some examples of real earthquakes showed that maximum accelerations for

vertical motion reached 80% of those for horizontal motion. In a few cases of shallow earthquakes, the vertical acceleration exceeded the horizontal acceleration.

W. NOVAK: As far as the rocking support is concerned, the vertical component is not considered. All the plant parts are examined to determine whether their natural frequencies are in the range of the earthquake vibrations with the greatest amplitudes. If these frequencies are in resonance, the accelerations are greater than can be expected in the case of the reference earthquake. If this additional loading should lead to unacceptable stresses consideration can be given to detuning, strengthening and damping.

E. IANSITI: How was the reference earthquake selected? Where were the strong motion records that you mentioned obtained, in your country or elsewhere in Europe? I ask this because the solution you use must be strongly dependent on the reference spectrum selected.

W. NOVAK: The reference earthquake was defined by geophysicists in collaboration with Swiss safety authorities. Assuming a rigid foundation, the maximum amplitude will not be more than ~ 0.3 mm. An amplitude of 1 mm causes a support movement of $< 1^\circ$, and the maximum permissible movement is 5.7° assuming a friction coefficient of $\sim 0.1\%$.

DISCUSSION ON THE FOLLOWING PAPER:

P/580 France Presented by O. Housseaux

There was no discussion on this paper.

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/543 CSSR Presented by J. Beránek

P/290 Norway Presented by E. Jamne

A. AFSHAR-BAKESHLOO: Mr. Beránek (P/543) mentioned the lack of water resources in the Czechoslovak Socialist Republic. I should like to ask whether use is made of cooling tower systems.

J. BERÁNEK: Yes.

A. LÉVAI: I have three questions to Mr. Beránek. First, I should like to know whether it has been definitely decided to build industrial or heating nuclear power plants in Czechoslovakia or are the examples you have given only studies?

J. BERÁNEK: It has been decided to build two nuclear power plants, each with two units of about 440 MW(e) during the next five years. The sites for these two nuclear power plants have already been approved — one in Jaslovské Bohunice, where the first nuclear power plant is also located, and the second in South Moravia. As far as nuclear heating or dual-purpose plants in densely populated areas are concerned, no decision has yet been made and all the information presented relates to design.

A. LÉVAI: What allowance is made for the cost of additional safety protection as compared with conventional power stations?

J. BERÁNEK: I have no economic data with me, but the cost of safety is not very important in comparison with the total investment cost, of which it forms an integral part.

A. LÉVAI: Is it intended to provide an ice condenser between the double walls of the containment, as proposed — I think — in an article by Westinghouse? Or are you satisfied with the normal double containment?

J. BERÁNEK: We are studying different types of double containment including the use of an ice condenser. For urban siting the use of ice condensers may have certain advantages.

P. POPA: I have two questions to ask Mr. Beránek. In the case of areas of low population density, if I understood correctly, you have established a 3-km protective zone on the basis of data for normal operation of the station. What do you think could be the radiological consequences of an accident for the area beyond the 3 km limit?

J. BERÁNEK: There is no need for the protective zone of radius 3 km under the conditions of normal operation. The protective area gives an additional factor of safety to allow for the case of accidental releases. Our safety analyses of possible emergency situations have convinced us that the risks associated with possible releases of radioactive material are acceptable. We are going to continue detailed evaluation on the basis of Safety Reports.

P. POPA: What hypothetical accident is taken as a basis in designing the containment for the Holešovice station?

J. BERÁNEK: The reference accident considered in the design study for the Holešovice nuclear power plant is double-end rupture of the main section of the primary circuit with immediate loss of coolant and subsequent core damage and release of 3.5-4% of the total radioiodine.

J. TADMOR: My question is to Mr. Jamne (P/290). The figure of 4% for the increased construction costs of an underground reactor as compared with a surface reactor seems relatively low. Figures given in previous studies vary between 10 and 20%. Would Mr. Jamne please comment on this difference. Is it due to specific Norwegian conditions or to other factors?

E. JAMNE: We arrived at the figure of approximately 4% after quite a detailed study covering the siting of each station on the surface and underground — at specific building sites. As pointed out in the paper, the cost difference will be strongly dependent on the sites selected. For your information I should perhaps mention that the total cost of the caverns and tunnels for these 500 MW(e) plants amounted to 2.6-2.8 million dollars (1969).

C. VELEZ: I should like to ask Mr. Jamne whether any consideration has been given to the possibility of reducing seismic accelerations by underground siting?

E. JAMNE: No. Seismic activity in our country is low, and we have not yet looked into this question.

C. O'FARRELLY: The underground stations appear to have an access tunnel backing directly from the reactor cavern to the outside. In the event of pressure-vessel rupture — and the capacity for coping with this is the main advantage of underground containment — don't you think there is a

possibility that a pressure wave could travel down the tunnel, demolish the airlock and so release the fission products to the atmosphere?

E. JAMNE: This problem was looked into in the Holden case. Figure 7 in the paper shows that the tunnel has been formed as an explosion trap between the hall and the airlock. It should be pointed out, however, that the MCA for this plant does not include vessel rupture. An important part of our future study will be to investigate the problems related to a vessel rupture in an underground plant. A main item here will of course be the protection of airlocks against shock waves.

AGENDA ITEM 3.4

Legislative, insurance and regulatory aspects

Problèmes de législation, de réglementation et d'assurance

Законодательные, страховые и регламентационные аспекты

Problemas de legislación, de seguros y de reglamentación

Chairman

D.'B. SOLE, South Africa

Vice-Chairman

J. SOEIRO DE BRITO, Portugal

Scientific Secretaries

J.D. McCULLEN, IAEA

H. EISENLOHR, IAEA

LA RESPONSABILITE CIVILE DES EXPLOITANTS D'INSTALLATIONS NUCLEAIRES ET L'ASSURANCE DE CETTE RESPONSABILITE

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Abstract—Résumé—Аннотация—Resumen

CIVIL LIABILITY OF THE OPERATORS OF NUCLEAR INSTALLATIONS AND INSURANCE AGAINST THIS LIABILITY.

Several international conventions have established a special system of third party liability for operators of nuclear installations. The rules of nuclear law ensure the protection of the public and workers, while allowing for the development of this new industry. The role of international organizations such as the IAEA, ENEA and EURATOM has been an important factor in the elaboration of the nuclear conventions. The four relevant Conventions are the Paris Convention on Third Party Liability in the Field of Nuclear Energy (in force since 1968), the Vienna Convention on Civil Liability for Nuclear Damage, the Brussels Convention on the Liability of Operators of Nuclear Ships, and the Convention of Brussels, supplementary to the Paris Convention. The important features of these Conventions are: (a) absolute and exclusive liability of the operator; (b) limited liability; and (c) insurance or financial security for this liability. The special system was intended to cover risks of an exceptional character that could not be covered appropriately by common law. It is important that definitions of nuclear installations and the scope of the Conventions should be harmonized. In countries whose legislation conforms with the nuclear conventions, there are no serious difficulties over insurance and national pools of nuclear insurers have in many cases been set up. Insurance may be either per installation per period or per incident, and carriage is normally covered separately from insurance of the installation. The nuclear conventions do not affect international transport conventions and there is thus an exception to the exclusive liability of a nuclear operator. As a result maritime carriers demand an unlimited indemnity from the operator before agreeing to carry, and this raises practical difficulties which a new draft maritime convention may help to overcome. A large number of countries have now achieved a measure of unification of nuclear law. Further ratification of the nuclear conventions would improve the situation still more. The nuclear conventions provide for unity of jurisdiction of courts dealing with claims and thus has advantages for victims.

LA RESPONSABILITE CIVILE DES EXPLOITANTS D'INSTALLATIONS NUCLEAIRES ET L'ASSURANCE DE CETTE RESPONSABILITE.

La responsabilité civile des exploitants d'installations nucléaires relève d'un régime spécial établi par diverses conventions internationales. Les principes du droit nucléaire sont tels qu'ils assurent la protection du public et des travailleurs tout en permettant le développement de cette nouvelle industrie. Les organisations internationales telles que l'AIEA, l'ENEA et EURATOM ont joué un rôle de premier plan dans l'élaboration des conventions nucléaires qui sont: la Convention de Paris sur la responsabilité civile dans le domaine de l'énergie nucléaire (en vigueur depuis 1968), la Convention de Vienne relative à la responsabilité civile en matière de dommages nucléaires, la Convention de Bruxelles relative à la responsabilité des exploitants de navires nucléaires et la Convention de Bruxelles complémentaire à la Convention de Paris. Ces conventions reposent sur les principes suivants: a) l'exploitant, et lui seul, est objectivement responsable; b) la responsabilité est limitée; et c) cette responsabilité est couverte par une assurance ou une garantie financière. L'objet du régime spécial est de couvrir des risques présentant un caractère exceptionnel, contre lesquels le droit civil ne pourrait assurer une protection satisfaisante. Il est essentiel d'harmoniser les définitions des installations nucléaires et le champ d'application des conventions. Dans les pays dont la législation est conforme aux conventions nucléaires l'assurance ne pose aucun problème sérieux, et des groupes nationaux d'assurance nucléaire se sont constitués. L'assurance peut couvrir soit une installation, soit une période, soit un accident et, en général, le transport est couvert par une assurance distincte de celle

de l'installation. Les conventions nucléaires ne modifient pas les conventions de transport international, ce qui crée une exception au principe selon lequel l'exploitant nucléaire est objectivement responsable. De ce fait, les transporteurs maritimes exigent que l'exploitant soit couvert par une garantie illimitée avant de consentir à effectuer un transport et cela donne lieu à des difficultés pratiques qui pourraient peut-être être résolues par une nouvelle convention maritime. Un grand nombre de pays sont maintenant parvenus à unifier dans une certaine mesure leur droit nucléaire. De nouvelles ratifications des conventions nucléaires amélioreraient encore la situation à cet égard. Les conventions nucléaires prévoient une unité de juridiction pour le règlement des différends, ce qui présente des avantages pour les victimes.

ГРАЖДАНСКАЯ ОТВЕТСТВЕННОСТЬ ЗА ЭКСПЛУАТАЦИЮ ЯДЕРНЫХ УСТАНОВОК И СТРАХОВАНИЕ ЭТОЙ ОТВЕТСТВЕННОСТИ.

Несколько международных конвенций определили специальную систему ответственности операторов ядерных установок перед третьей стороной. Положения законодательства в области использования ядерной энергии обеспечивают защиту прав населения и работающих в области использования ядерной энергии и в то же время не препятствуют развитию этой новой отрасли промышленности. В разработке конвенций по ядерной энергии большую роль сыграли такие международные организации, как МАГАТЭ, ЕАЯЭ и ЕВРАТОМ. Ядерное право возникло на основе четырех конвенций: Парижской конвенции об ответственности перед третьей стороной в области ядерной энергии (действует с 1968 года), Венской конвенции о гражданской ответственности за ядерный ущерб, Брюссельской конвенции об ответственности операторов ядерных судов и Брюссельской конвенции, дополняющей Парижскую конвенцию. Наиболее важные положения этих конвенций охватывают следующее: (а) абсолютную и исключительную ответственность оператора; (б) ограниченную ответственность и (с) страхование или финансовое обеспечение этой ответственности. Специальная система предусматривает покрытие риска исключительного характера, который не может быть соответствующим образом охвачен положениями общего права. Важно, чтобы определения ядерных установок и положения конвенции соответствовали друг другу. В странах, где законодательство учитывает конвенции по ядерному праву, не возникает серьезных трудностей в вопросах страхования, а в ряде стран созданы даже национальные объединения ядерных страховщиков. Страхование может производиться по отдельным установкам, по срокам или по случаям аварий, а страхование транспортировки обычно производится отдельно от страхования установки. Ядерные конвенции не распространяются на международные транспортные конвенции, и поэтому существует исключение в ответственности ядерного оператора. В результате этого морские перевозчики требуют от оператора неограниченной гарантии возмещения ущерба, прежде чем они согласятся на перевозку. Это создает практические трудности, которые новый проект морской конвенции, вероятно, поможет преодолеть. Большое число стран в настоящее время добилось значительной унификации законодательства в области использования ядерной энергии. Дальнейшая ратификация ядерных конвенций еще более улучшит это положение. Ядерные конвенции предусматривают единство юрисдикции судов, в которых рассматриваются претензии. Это даст определенные преимущества для потерпевших сторон.

RESPONSABILIDAD CIVIL DE LOS EXPLOTADORES DE INSTALACIONES NUCLEARES Y SEGURO PARA CUBRIR ESA RESPONSABILIDAD.

En diversos convenios y convenciones internacionales se ha establecido un sistema especial de responsabilidad civil que afecta a los explotadores de las instalaciones nucleares. Las normas del derecho nuclear aseguran la protección del público y de los trabajadores, a la vez que permiten el desarrollo de esta nueva industria. La actuación de las organizaciones internacionales, tales como el OIEA, la AEA y la EURATOM ha constituido un factor importante en la elaboración de estas convenciones y convenios nucleares. Son éstos cuatro: el Convenio de París acerca de la Responsabilidad Civil en Materia de Energía Nuclear (en vigor desde 1968), la Convención de Viena sobre Responsabilidad Civil por Daños Nucleares, la Convención de Bruselas sobre la Responsabilidad de los Explotadores de Buques Nucleares y el Convenio de Bruselas Complementario del Convenio de París. Las características importantes de estos tratados son las siguientes: a) la responsabilidad objetiva y exclusiva del explotador; b) la responsabilidad limitada y c) el seguro u otra garantía financiera que cubra esta responsabilidad. Con este sistema especial se pretendían cubrir los riesgos de carácter excepcional que no quedarán adecuadamente cubiertos por el derecho común. Procederá armonizar las definiciones de las instalaciones nucleares y los campos de aplicación de los tratados (convenciones y convenios). En aquellos países cuya legislación se ajusta a estos tratados nucleares no existen graves dificultades en materia de seguro y se han establecido en muchos casos consorcios nacionales de aseguradores nucleares. El seguro puede ser por instalación, por período de tiempo o por accidente, y el transporte se cubre normalmente por separado del seguro de la instalación. Los tratados nucleares no afectan a las convenciones de transporte internacional y, por tanto, se presenta aquí un caso de excepción de la responsabilidad exclusiva del explotador nuclear. Como consecuencia, las empresas de transporte marítimo

exigen que el explotador las declare plenamente exentas de responsabilidad antes de concertar el transporte, lo que suscita dificultades prácticas que un nuevo proyecto de convención marítima podría contribuir a superar. Un gran número de países han logrado ya una cierta unificación en materia de legislación nuclear. Una nueva ratificación de las convenciones y convenios nucleares permitiría mejorar más aún la situación. Estos tratados nucleares establecen la unidad de jurisdicción de los tribunales competentes en materia de demandas, lo que redundará en beneficio de las víctimas.

1. Avant même que l'énergie nucléaire ne soit entrée dans une phase de développement industriel, plusieurs conventions internationales, conclues entre 1960 et 1963, ont établi un régime de responsabilité civile des exploitants d'installations nucléaires qui est fondé sur des bases tout à fait différentes de celles du Droit traditionnel. Ce phénomène est important du point de vue de l'évolution juridique.

La plupart des règles du droit nucléaire ont été conçues, a priori, pour assurer la protection du public et des travailleurs grâce à des précautions, des contrôles et des garanties de nature exceptionnelle, tout en permettant le développement satisfaisant d'une nouvelle branche d'industrie. Ces règles n'ont pas été imposées par la nécessité de faire face à un risque immédiat et n'ont pas été influencées par des intérêts acquis. Alors que bien souvent le droit s'efforce, avec un certain retard, d'ordonner et de contrôler des situations de fait qui sont bien connues mais qu'il est souvent difficile de modifier, le droit nucléaire s'est formé avec une certaine avance sur les faits. Il devra donc subir l'épreuve de l'expérience pratique.

En deuxième lieu, le régime de la responsabilité civile nucléaire offre l'exemple rare d'un ensemble de dispositions de droit privé qui a été entièrement élaboré sur le plan international. Ceci entraîne tous les avantages que présente l'unification du droit pour les relations entre les pays, mais comporte aussi les difficultés propres à l'incorporation des conventions internationales dans les droits nationaux. Il faut souligner ici le rôle des grands organismes de coopération qui ont servi de cadre à la négociation des conventions nucléaires, c'est-à-dire l'Agence internationale de l'énergie atomique (AIEA), l'Agence européenne pour l'énergie nucléaire (ENEA) et la Communauté européenne de l'énergie atomique (Euratom). Leur action en faveur d'une harmonisation législative se poursuit tout naturellement jusqu'au stade de l'application effective des règles internationales.

Enfin, nous sommes en présence de règles juridiques originales qui ont été conçues en fonction d'une technique nouvelle, pour tenir compte du caractère spécifique des dommages susceptibles d'être causés par la mise en œuvre de la fission et de la désintégration de l'atome, dommages qui ne sont pas seulement caractérisés par leur ampleur potentielle mais surtout par la difficulté d'apporter la preuve de leur origine et par l'effet différé des radiations sur l'organisme humain. La nouveauté de ces règles suscite inévitablement la résistance des principes traditionnels de responsabilité, qui gouvernent la plupart des activités humaines et qui correspondent à une phase antérieure de l'évolution technique.

2. Nous sommes donc en présence d'une situation quelque peu ambiguë. Les avantages du nouveau régime de responsabilité, qui découlent de son « adéquation » à une technique moderne et de son origine internationale,

sont également la cause des difficultés et des lenteurs que l'on observe dans la mise en application effective de ce régime. Il est cependant encourageant de constater que ces avantages semblent s'imposer très largement car les législations de nombreux pays qui n'ont pas encore ratifié les conventions nucléaires s'inspirent cependant très largement de celles-ci.¹ Les inconvénients dus aux retards dans la ratification des conventions sont d'ailleurs provisoirement atténués par le fait que le développement industriel de l'énergie nucléaire est moins rapide que prévu.

3. Nous ne consacrerons dans le présent exposé qu'une place limitée à la description des règles relatives à la responsabilité civile nucléaire, qui sont maintenant relativement bien connues. Nous concentrerons plutôt notre attention sur quelques problèmes pratiques liés à la mise en place de ce nouveau cadre juridique et sur les aspects internationaux de la responsabilité des exploitants d'installations nucléaires.

I. LE CONCEPT DE RESPONSABILITE CIVILE NUCLEAIRE DANS LES CONVENTIONS INTERNATIONALES

4. L'ensemble du régime de la responsabilité civile nucléaire a été établi par quatre conventions internationales dont seule la première, par ordre chronologique, est actuellement en vigueur. Ces conventions sont les suivantes:

- la Convention de Paris sur la responsabilité civile dans le domaine de l'énergie nucléaire, qui a été signée le 29 juillet 1960 par seize pays d'Europe occidentale et qui est en vigueur depuis le 1^{er} avril 1968;
- la Convention de Vienne sur la responsabilité civile pour les dommages nucléaires, qui a été adoptée le 21 mai 1963 et qui a pour objet d'établir sur le plan mondial un régime analogue à celui de la Convention de Paris;²
- la Convention de Bruxelles relative à la responsabilité des exploitants de navires nucléaires, adoptée le 25 mai 1962, dont l'objet est de fournir pour les navires nucléaires un régime fondé sur les mêmes principes que celui des installations fixes;
- la Convention complémentaire à la Convention de Paris, qui a été signée à Bruxelles le 31 janvier 1963 pour apporter des possibilités supplémentaires de réparation des dommages nucléaires, grâce à une intervention financière des gouvernements.³

Un état des signatures, ratifications et accessions, pour les quatre conventions, est donné en annexe.

5. Nous nous contenterons de rappeler les grandes lignes de ces conventions, en laissant de côté les dispositions qui sont d'importance secondaire.

¹ Sur les législations nationales des pays de l'OCDE, voir le fascicule « Responsabilité civile » (1967) dans la série « Législations nucléaires » publiée par l'ENEA, ainsi que les numéros 1 à 7 (1968 à 1971) du Bulletin de droit nucléaire, également publié par l'ENEA.

² Un protocole additionnel à la Convention de Paris, signé le 28 janvier 1964, a pratiquement aligné les dispositions de cette Convention sur celle de Vienne en supprimant les différences qui existaient entre les deux textes; celles qui subsistent sont d'une importance pratique limitée ou d'ordre rédactionnel.

³ Dans le présent exposé, on se référera à ces conventions de la manière suivante: Paris, Vienne, Bruxelles et Complémentaire de Bruxelles. Le texte de ces conventions figure dans le n°4 de la Collection juridique de l'AIEA (1966).

A. L'exploitant d'une installation nucléaire est responsable objectivement et exclusivement pour tous les dommages (couverts par les conventions) qui sont causés par un accident nucléaire survenu dans cette installation ou qui mettent en jeu des matières (ou substances) nucléaires, en cours de transport, qui proviennent de cette installation ou y sont envoyées (Paris art. 3 et 4; Vienne art. II). L'exploitant d'un navire nucléaire est responsable, dans les mêmes conditions, pour les accidents où sont impliquées les matières nucléaires de ce navire (Bruxelles art. II). Les principales conséquences de cette règle sont les suivantes:

- il n'est jamais nécessaire de prouver la faute de l'exploitant;
- l'exploitant ne peut jamais se décharger de sa responsabilité en prouvant qu'il n'a pas commis de faute ou qu'un tiers a commis une faute;
- personne d'autre que l'exploitant ne peut être rendu responsable.

Cette règle de la « canalisation » de la responsabilité, que nous venons d'énoncer, ne comporte que peu d'exceptions ou d'atténuations et celles-ci sont énumérées de manière limitative:

- exception à la responsabilité de l'exploitant en cas de dommages résultant de conflits armés, de guerres civiles, d'insurrections ou de cataclysmes naturels de caractère exceptionnel (Paris art. 9; Vienne art. IV. 3); pour les navires, les cataclysmes naturels ne sont par prévus parmi les causes d'exonération de responsabilité (Bruxelles art. VIII); il faut remarquer que de toute manière les exceptions prévues par les trois conventions sont plus limitées que la notion habituelle de force majeure;
- exception à la règle de la responsabilité exclusive de l'exploitant pour les dommages survenus à l'occasion de transports internationaux (cf. paragraphe 16 ci-dessous);
- atténuation à la règle de la responsabilité objective de l'exploitant par la faculté laissée aux législations nationales de diminuer ou de supprimer l'obligation de réparation envers la victime d'un dommage lorsque cette victime a contribué intentionnellement ou par sa faute grave à la cause de ce dommage (Paris, implicitement art. 11 et 1^{er} Exposé des motifs, paragraphes 48 et 52; Vienne art. IV. 2; Bruxelles art. II. 5);
- atténuation à la règle de la responsabilité exclusive par l'attribution à l'exploitant d'un droit de recours envers une personne qui a causé intentionnellement un dommage nucléaire ou lorsqu'un tel recours a été expressément prévu par contrat (Paris art. 6f; Vienne art. X; Bruxelles art. II. 6).⁴

Il y a lieu de remarquer que ces différentes exceptions et atténuations à la règle générale (sauf celles relatives aux transports internationaux) ne concernent que des hypothèses dont la probabilité est très faible et qu'elles ne conduisent jamais à reconnaître la notion de faute comme fondement de la responsabilité de l'exploitant nucléaire ou d'une responsabilité directe d'une autre personne vis-à-vis des victimes d'un dommage couvert par les conventions.

B. La responsabilité de l'exploitant d'une installation ou d'un navire nucléaire est limitée à un montant maximal fixé soit par la convention

⁴ Un troisième cas de recours existe au profit de l'exploitant de navire nucléaire, dans certaines circonstances, pour un accident nucléaire résultant de travaux de relèvement de l'épave.

internationale applicable, soit par la loi nationale conformément à la Convention. Pour les installations fixes, ce montant ne peut jamais être inférieur à 5 millions de dollars des Etats-Unis par accident et il est en principe de 15 millions de dollars pour les pays signataires de la Convention de Paris (Paris art. 7b; Vienne art. V.1); pour les navires nucléaires, ce montant est de 1,5 milliard de francs or de 1928, soit environ 100 millions de dollars, par accident (Bruxelles art. III.1 et 4).

La responsabilité des exploitants est également limitée dans le temps par des dispositions qui sont propres aux dommages nucléaires et qui tiennent compte de l'effet différé des radiations. En règle générale, les délais de prescription prévus par les conventions nucléaires sont de dix ans (Paris art. 8; Vienne art. VI; Bruxelles art. V).

C. L'exploitant d'une installation ou d'un navire nucléaire est tenu de souscrire et de maintenir une assurance ou une autre garantie financière pour couvrir sa responsabilité; son montant doit donc être, en principe, égal au montant maximal de cette responsabilité (Paris art. 10a; Vienne art. VII.1; Bruxelles art. III.2). Les fonds ainsi mis à disposition ne peuvent être utilisés que pour les dommages couverts par les Conventions (Paris art. 10c; Vienne art. VII.3; Bruxelles art. IX).

6. Il y a dans le concept de la responsabilité civile nucléaire une logique interne qui s'oppose assez fortement à la dissociation des différents éléments qui la composent. La responsabilité objective et l'obligation d'assurance sont évidemment fondées sur les exigences de la protection des victimes; la responsabilité exclusive est une conséquence inévitable de l'abandon de la notion de faute, mais elle est également liée à l'obligation d'assurance, puisque l'existence d'un seul responsable permet d'assurer le risque dans de meilleures conditions (cf. paragraphe 14 ci-dessous); la limitation de responsabilité est imposée, d'une part, par le souci de ne pas empêcher le développement de l'énergie nucléaire – comme une contrepartie de la responsabilité objective – et, d'autre part, par l'obligation d'assurance qui dans le domaine nucléaire ne pourrait pas porter sur une couverture illimitée.

7. La limitation de la responsabilité de l'exploitant conduit, en outre, à se poser le problème de la réparation des dommages nucléaires excédant le montant maximal de cette responsabilité, et cela d'autant plus que ce montant est moins élevé. On peut alors se demander si la prise en charge de ces dommages par le gouvernement du pays où se trouve l'installation n'est pas le prolongement naturel du régime que nous avons décrit. La Convention de Vienne ne contient pas de dispositions à cet égard et la Convention sur les navires nucléaires se borne à prendre des précautions relatives à la répartition des fonds disponibles dans l'hypothèse d'un dépassement probable (Bruxelles art. XI). Cependant, la Convention de Paris prévoit (art. 15) la possibilité pour les Etats signataires de prendre des mesures particulières en vue d'accroître l'importance de la réparation, notamment au moyen de fonds publics. Dans cet esprit, treize des pays signataires de la Convention de Paris ont conclu la Convention complémentaire de Bruxelles qui oblige les gouvernements à réparer les dommages nucléaires, au-delà du montant de la garantie financière de l'exploitant, jusqu'à

concurrence de 120 millions de dollars.⁵ Enfin, il existe à l'heure actuelle de nombreuses législations nationales (en Europe, aux Etats-Unis, au Canada, au Japon...) qui prévoient, de manière plus ou moins précise et large, l'intervention financière des gouvernements pour compléter la responsabilité de l'exploitant.

8. Il y a une catégorie importante de dommages nucléaires auxquels les Conventions que nous avons évoquées ne s'appliquent pas. Il s'agit des dommages aux biens qui se trouvent sur le site de l'installation et qui sont utilisés pour son exploitation (Paris art. 3 a.ii) 1; Vienne art. IV.5), ainsi que des dommages subis par le navire nucléaire lui-même, ses agrès et appareils... (Bruxelles art. II.3). Il est normal que la propriété de l'exploitant responsable ne soit pas couverte par le régime de responsabilité civile, mais les dispositions de la Convention vont un peu plus loin puisque certains des biens visés peuvent être la propriété de tiers. Cette mesure est sans doute justifiée, d'une part par un souci de simplification, qui conduit à laisser dans le domaine de l'assurance-dommage et des relations contractuelles tous les dommages matériels sur le site, et d'autre part par la volonté de réserver le montant de la garantie financière aux victimes innocentes des activités de l'industrie nucléaire.

Dans le même ordre d'idées, les dommages aux moyens de transport employés pour les matières nucléaires n'entrent pas, en principe, dans le champ des Conventions, mais peuvent y être introduits par le législateur national à condition qu'un montant de 5 millions de dollars soit réservé pour les autres dommages aux tiers (Paris art. 3 a.ii) 2 et 7 c; Vienne art. IV.5 b et IV.6).

II. ASPECTS PRATIQUES DU REGIME DE RESPONSABILITE CIVILE NUCLEAIRE

Champ d'application technique

9. Il est un premier problème qui intéresse aussi bien l'industrie nucléaire que les autorités administratives: c'est celui de la délimitation du domaine technique dans lequel doit s'appliquer le régime de responsabilité civile qui vient d'être décrit.

Il est essentiel de savoir avec précision quelles sont les catégories d'installations qui doivent être soumises à ces règles, car nous sommes en présence d'un régime exceptionnel, justifié par le caractère spécifique et l'ampleur du risque créé pour les tiers par l'énergie nucléaire. L'Exposé des motifs de la Convention de Paris précise clairement que les dispositions de la Convention ne doivent s'étendre qu'aux risques de caractère exceptionnel auxquels les règles du droit commun de la responsabilité ne peuvent être appliquées sans inconvénient. Il en résulte évidemment que certaines activités qui impliquent l'utilisation de matières

⁵ Une première tranche est couverte par la garantie de l'exploitant; une deuxième tranche, jusqu'à 70 millions de dollars, par le gouvernement de l'Etat de l'installation; la troisième tranche (entre 70 et 120 millions de dollars) par l'ensemble des gouvernements signataires selon une clé de répartition fixée en tenant compte du produit national brut et de la puissance thermique des réacteurs de chaque pays (Complémentaire de Bruxelles art. 3 et 12).

radioactives ou même fissiles ne doivent pas être soumises à un régime spécial lorsque les règles habituelles fournissent des moyens suffisants pour faire face aux conséquences dommageables de ces activités. Il s'agit notamment d'activités qui ne mettent en jeu que de petites quantités de matière nucléaire dont le niveau de radioactivité reste peu élevé et qui ne sauraient justifier à elles seules la mise en place d'une responsabilité objective assortie d'une couverture financière obligatoire, d'un montant élevé.

10. Il est facile de comprendre que la frontière entre les installations nucléaires soumises à un régime exceptionnel et les installations qui doivent relever du droit commun sera, dans une certaine mesure, arbitraire. Il n'existe pas en fait de moyen d'évaluer avec précision l'ampleur du risque potentiel lié à une catégorie déterminée d'installations, car ce risque dépendra beaucoup des conditions locales particulières à chaque installation; une telle situation ne doit cependant pas décourager les autorités compétentes car les limites qu'il est nécessaire de prévoir pour l'application de dispositions légales sont presque toujours arbitraires.

Il est avant tout important pour l'industrie que le champ d'application de la responsabilité civile nucléaire soit déterminé avec précision afin que les exploitants connaissent les conditions dans lesquelles ils peuvent être rendus responsables et surtout les conditions dans lesquelles ils doivent assurer leur responsabilité. Ceci est également important pour les tribunaux qui auront à juger les réclamations des victimes de dommages liés à des matières nucléaires.

Enfin, du point de vue des relations internationales, il faut que les définitions des installations nucléaires soumises aux conventions internationales soient concordantes, en pratique, dans tous les pays signataires de ces conventions. Par exemple, en matière de transport de matières radioactives entre deux installations situées dans des pays différents, il est important de savoir, pour déterminer les responsabilités en cas de dommages survenus pendant le transport, si ces installations sont classées ou non comme installations nucléaires du point de vue des conventions.

On peut d'ailleurs remarquer que cette classification ne coïncide pas nécessairement avec celle qui aura été établie pour l'application du régime d'autorisation et de protection contre les radiations, bien que des considérations d'ordre administratif et technique militent en faveur de cette solution.

11. Les Conventions de Paris (art. 1) et de Vienne (art. I) contiennent des définitions identiques et relativement claires des installations nucléaires auxquelles elles s'appliquent. Cependant, il était impossible et il aurait été inopportun, dans ces documents juridiques, de donner une description technique détaillée des installations nucléaires visées; il a fallu se contenter de définitions générales qui peuvent évidemment donner lieu à quelques problèmes d'interprétation.

Les catégories d'installations nucléaires auxquelles s'appliquent les Conventions de Paris et de Vienne sont les réacteurs, les usines de fabrication ou de traitement de matières nucléaires, les usines de séparation isotopique et celles de traitement chimique de combustibles irradiés; les installations de stockage de matières nucléaires sont également visées mais non pas les laboratoires de recherche proprement dits.

Les deux Conventions s'appliquent également aux transports de matières fissiles ou radioactives, à l'exclusion des transports d'uranium naturel.

Par contre, elles ne s'appliquent pas aux radioisotopes destinés à des utilisations scientifiques, médicales, agricoles, industrielles ou commerciales, qui se trouvent en dehors d'une installation nucléaire, même lorsqu'elles sont en cours de transport entre de telles installations (Paris art. 1 a iv) et interprétation approuvée par le Comité de Direction de l'ENEA en avril 1968; Vienne art. I. 1 g).

12. L'expérience, ainsi que l'évolution scientifique et industrielle, sont susceptibles d'apporter des éléments nouveaux dont les définitions données par les Conventions n'auraient pas tenu compte. C'est pourquoi un mécanisme a été prévu qui permet, au moyen de décisions du Comité de Direction de l'ENEA pour la Convention de Paris (art. 1 b), et du Conseil des gouverneurs de l'AIEA pour la Convention de Vienne (art. I. 2), de restreindre le champ d'application technique des Conventions en excluant de petites quantités de matières nucléaires – ou même pour la Convention de Paris certaines catégories d'installations ou de matières – en raison des risques réduits qu'elles comportent. La Convention de Paris prévoit en outre la faculté d'élargir ce champ d'application, en allongeant, par le même procédé, la liste des installations nucléaires.

L'AIEA et l'ENEA ont été amenées, assez tôt, à aborder l'étude des exclusions qu'il paraissait opportun de proposer, en s'appuyant sur des considérations techniques. Une première série de décisions ont été prises pour exclure du champ d'application des Conventions de Vienne et de Paris de petites quantités de matières nucléaires en cours de transport.⁶ Des propositions en vue d'exclure également de petites quantités de matières⁷ lorsqu'elles se trouvent dans des installations qui ne sont pas autorisées à en contenir davantage ont fait l'objet d'examens approfondis au sein de l'AIEA et de l'ENEA. Lors de sa troisième série de réunions tenues du 1^{er} au 4 juin 1971, le Comité permanent de la responsabilité civile en matière de dommages nucléaires a formulé une recommandation dans ce sens, laquelle a été transmise au Directeur général de l'AIEA.

13. Les dispositions des Conventions relatives aux catégories d'installations et de matières nucléaires sont assez larges et les décisions tendant à restreindre le champ d'application des Conventions restent, bien entendu, très prudentes. On peut cependant voir, dans les unes et les autres, la volonté de laisser en dehors du régime de la responsabilité civile nucléaire des activités qui ne justifient pas un traitement exceptionnel. Ceci conduit, dans les cas limites, à exercer un jugement quelque peu délicat où des considérations d'ordre sociologique ont autant d'importance que l'analyse purement technique du risque. La portée de ce problème ne doit pas être exagérée, car dans la très grande majorité des cas il est facile de décider si une installation doit être considérée comme nucléaire ou pas et il en est notamment ainsi pour les réacteurs de puissance et les usines liées au cycle de l'uranium, qui constituent les unités essentielles du point de vue industriel.

⁶ Résolution du Conseil des gouverneurs de l'AIEA en date du 11 septembre 1964 et décision analogue du Comité de Direction de l'ENEA du 26 novembre 1964 (voir Collection juridique de l'AIEA, n°4, 1966).

⁷ Selon les projets des experts de l'ENEA, il devrait, pour certaines matières, s'agir de quantités plus importantes que celles qui ont été fixées en 1964 pour le transport.

Pour terminer, il faut ajouter que dans l'état actuel de la technique la définition technique des navires nucléaires auxquels s'applique la Convention de Bruxelles de 1962 ne semble poser aucun problème: il s'agit de tous les navires équipés d'un réacteur de puissance, utilisé pour la propulsion ou à d'autres fins (art. I. 1 et 3), ce qui ne laisse pas de place pour le doute.

Assurance de la responsabilité civile

14. Un des intérêts pratiques essentiels du régime de responsabilité civile pour les installations nucléaires est de fournir une base juridique satisfaisante pour l'assurance des dommages causés aux tiers. Les assureurs qui ont accepté au départ l'idée de couvrir un tel risque ont d'ailleurs joué un rôle actif dans l'étude et la mise en place de ce régime. A l'heure actuelle, l'assurance de la responsabilité civile ne semble pas poser de problèmes graves ni pour l'industrie nucléaire, ni pour les assureurs, dans les pays qui ont une législation conforme aux Conventions de Paris ou de Vienne.

Il est difficile de porter un jugement critique sur le niveau des primes d'assurance qui sont à la charge de l'industrie nucléaire par rapport au niveau qui est appliqué dans d'autres secteurs industriels. Ces primes dépendent en effet, dans une grande proportion, des conditions locales particulières à chaque installation. Il faut en outre tenir compte, d'une part de certains coûts propres à l'industrie nucléaire, comme par exemple les coûts de décontamination, et d'autre part du fait que la police de l'exploitant couvre l'ensemble des responsabilités canalisées sur lui, car dans un régime de droit commun elles feraient l'objet de polices multiples. Cette concentration de l'assurance, qui est le reflet de la canalisation juridique, contribue également à simplifier les relations contractuelles entre l'exploitant et les entreprises qui fournissent des services à l'installation et dont une faute pourrait être à l'origine d'un dommage à des tiers: bureaux d'études, fournisseurs de travaux, d'équipements ou de combustibles, transporteurs, etc. La responsabilité exclusive de l'exploitant décharge ces fournisseurs d'une responsabilité propre qu'ils n'auraient pas pu accepter car elle dépasse leur risque professionnel habituel. Cette situation conduit logiquement les assureurs et l'exploitant à abandonner des recours contractuels qui n'ont pas d'intérêt économique et qui, de toute manière, se heurteraient à de fortes résistances. Il est probable qu'en évitant ainsi la multiplication des polices on réalise une économie sur le montant global des primes.

La suppression des polices multiples conduit également à une réduction des garanties financières à mobiliser pour couvrir l'ensemble du risque nucléaire pour les tiers, qui sont de toute manière plus élevées que dans d'autres branches d'industries. Le risque étant le même quelles que soient les modalités d'assurance, il serait artificiel d'accorder, pour le même dommage, des couvertures considérables à plusieurs responsables potentiels.

Le développement de la production d'énergie nucléaire entraînera une augmentation rapide du nombre d'installations à couvrir; dès maintenant, on peut estimer que le nombre d'installations dans un pays européen qui a un programme nucléaire important dépasse la cinquantaine. Pour faire face à cette demande, les assureurs ont établi des pools nationaux

d'assurance nucléaire ainsi qu'une coopération sur le plan international, qui facilitent la mise en œuvre des techniques de coassurance et de réassurance sur une large échelle. Cette organisation s'appuie sur la politique adoptée par les gouvernements lorsqu'ils ont décidé d'instituer, sur le plan international, un régime spécial de responsabilité pour l'industrie nucléaire.

15. La couverture de la responsabilité pour les dommages nucléaires soulève un certain nombre de questions particulières qu'il est intéressant de mentionner.⁸

Le point de départ de la responsabilité de l'exploitant et par conséquent de la police d'assurance correspondante n'est pas fixé par les Conventions. L'exploitant est, en vertu de ces Conventions, la personne désignée par l'autorité compétente et le contrôle de l'obligation d'assurance est le plus souvent lié à la procédure d'autorisation des installations. C'est donc aux règlements et aux autorités nationales qu'il appartient de déterminer le moment où la responsabilité objective et exclusive de l'exploitant commence. On peut se demander si ce moment doit coïncider avec la présence de matières fissiles ou radioactives dans l'installation, même pendant la période des essais, ou s'il faut attendre la mise en exploitation, ce qui aurait l'inconvénient de faire se succéder deux régimes de responsabilité différents pour des dommages nucléaires.

L'exploitant peut également être responsable pour des dommages purement classiques qui ne sont pas liés à un accident nucléaire, mais la tendance générale des assureurs en Europe est de les couvrir dans la même police d'assurance que les dommages nucléaires, ceci dans un but de simplification; il est bien entendu que le régime juridique des dommages classiques n'en sera pas modifié pour autant.

Nous avons vu que l'exploitant est également responsable, en vertu des conventions nucléaires, des dommages survenus en cours de transports. Dans la plupart des pays, les transports font l'objet de polices d'assurance qui sont différentes de celle de l'installation, mais il est également possible de couvrir ces deux catégories de risques dans la même police. Dans les deux cas, les assureurs doivent fournir un certificat qui a pour objet d'établir la consistance de l'assurance qui a été souscrite et qui présente une utilité pratique pour les transports internationaux. Le Comité de Direction de l'ENEA a recommandé aux pays signataires de la Convention de Paris un modèle de certificat qui sera reconnu comme valable par les autorités compétentes des pays intéressés.⁹

Le problème le plus délicat concerne le maintien de l'intégrité de la couverture après la survenance d'un accident. Les conventions nucléaires ou les lois d'application imposent un montant d'assurance déterminé pour chaque accident. Toutefois, au stade actuel, en l'absence de statistiques significatives sur la probabilité des accidents nucléaires, les assureurs estiment qu'ils sont seulement en mesure de garantir ce montant maximal pour chaque installation pendant la durée de la police et non pas pour chaque accident. Le montant de la couverture s'épuise donc après chaque accident et il est nécessaire de reconstituer la garantie jusqu'à concurrence

⁸ Pour une étude détaillée, voir *Nuclear Law for a Developing World* (AIEA, Collection juridique n°5, 1969) et *Insurance for Nuclear Installations* (AIEA, Collection juridique n°6, 1970).

⁹ Voir ce modèle dans le 9^e Rapport d'activité de l'ENEA. Annexe B.

du montant légal. Ceci peut entraîner des inconvénients pour la continuité d'une exploitation ou la poursuite d'un transport et entraîner pour l'exploitant le paiement de primes supplémentaires. Il faut espérer que l'expérience permettra aux assureurs de modifier leur politique à ce sujet.

Transport maritime de substances nucléaires

16. Comme nous l'avons indiqué, les Conventions de Paris et de Vienne contiennent une exception à la règle de la canalisation juridique dans le domaine des transports internationaux. Les rédacteurs de ces Conventions n'ont pas voulu en effet porter atteinte aux nombreux accords internationaux qui régissent ces transports et qui étaient en vigueur ou en cours de signature et de ratification à ce moment-là; ces accords continuent donc à s'appliquer (Paris art. 6; Vienne art. II, 5). Cette exception a soulevé des problèmes en ce qui concerne le transport maritime.

En cas de dommages nucléaires survenant pendant un transport maritime, la situation sera en résumé la suivante: dans la mesure où la Convention de Paris ou de Vienne est applicable, l'exploitant de l'installation nucléaire en cause sera objectivement responsable en vertu de l'une ou l'autre de ces Conventions, mais en outre l'armateur (ou une autre personne) pourra être rendu responsable pour ces dommages en vertu d'une convention maritime et il est possible que cette responsabilité soit illimitée.¹⁰ C'est pourquoi les armateurs hésitent beaucoup à se charger du transport de substances nucléaires et demandent le plus souvent une garantie financière illimitée pour les conséquences des dommages nucléaires qui ne seraient pas couverts par l'assurance de l'exploitant nucléaire responsable. L'expérience montre qu'il en résulte des difficultés pratiques considérables.¹¹

Afin de surmonter ces difficultés, l'AIEA et l'ENEA, en collaboration avec le Comité maritime international et le Comité européen des assurances, ont élaboré un projet de nouvelle convention maritime qui aurait pour objet d'exonérer les armateurs (ou autres personnes) de leur responsabilité pour les dommages nucléaires en vertu du droit maritime international ou national, chaque fois qu'un exploitant nucléaire serait responsable pour ces dommages en vertu de la Convention de Paris ou de Vienne. Ce projet est actuellement en cours d'examen par l'Organisation intergouvernementale consultative pour la navigation maritime, et il est permis d'espérer qu'une solution satisfaisante interviendra bientôt.

III. LES EFFETS INTERNATIONAUX DES CONVENTIONS NUCLEAIRES

17. Il est important d'examiner les aspects internationaux du régime unifié de la responsabilité civile. Comme nous l'avons dit, une certaine unification du droit nucléaire a été réalisée grâce à la coopération internationale et

¹⁰ Nous laissons de côté le problème soulevé par l'application de la législation nationale d'un pays qui n'a pas ratifié la Convention de Paris ou de Vienne. Les Conventions maritimes susceptibles de s'appliquer sont la Convention de 1910 sur l'abordage, la Convention de 1924 sur le connaissement, et les Conventions de 1924 et 1957 sur la limitation de responsabilité des propriétaires de navires.

¹¹ Une étude complète des problèmes juridiques et d'assurance est fournie par le compte rendu du Colloque de Monaco sur le transport maritime de substances nucléaires qui a été organisé conjointement par l'ENEA et l'AIEA en octobre 1968 (publ. en 1970).

les résultats des travaux effectués pour élaborer les Conventions ont fortement inspiré les législateurs de nombreux pays. L'unification des règles de responsabilité civile nucléaire est en elle-même de nature à faciliter, dans une certaine mesure, la réparation des dommages causés par un accident survenu à l'occasion d'activités qui mettent en cause les intérêts de plusieurs pays. Cependant, la portée internationale de cette unification «extraconventionnelle» reste limitée puisque les règles de responsabilité adoptées sur le plan national, même si ces règles sont identiques dans tous les pays intéressés, ne créent pas pour autant d'obligations internationales réciproques. En outre, les problèmes relatifs à la compétence des tribunaux, à la loi applicable et à l'exécution des jugements à l'étranger ne peuvent être résolus efficacement que par la voie des conventions internationales. Il y a donc lieu d'examiner quels sont les avantages supplémentaires que peut apporter la ratification des conventions qui ont été signées.

Remarquons tout d'abord que le commerce international est amené à tenir une place importante dans un domaine qui nécessite la mise en œuvre de techniques très spécialisées, la construction d'installations de grande envergure dont tous les pays ne pourront pas disposer, ainsi que l'emploi de matières très coûteuses qui ne seront produites que dans un nombre limité de pays. Les transactions internationales seront donc nombreuses. Des problèmes de responsabilité civile se poseront notamment pour l'étude de projets de centrales ou d'usines nucléaires pour des pays étrangers, la fourniture d'équipements ou de matières nucléaires pour ces réacteurs ou ces installations, ainsi que les transports de matières nucléaires d'un pays à un autre. L'expérience a montré que ces problèmes de responsabilité étaient difficiles à régler et il est arrivé que la construction d'installations nucléaires ait été retardée ou même empêchée par l'absence de législation ou d'accords internationaux appropriés.

18. Le premier effet des ratifications sera d'éliminer les divergences d'importance pratique qui existent encore entre les législations nationales ainsi que les risques d'une modification unilatérale de ces législations qui pourrait par exemple être défavorable à des fournisseurs étrangers. Il faut ajouter que dans un litige concernant deux pays qui ont ratifié une convention nucléaire, cette convention aura une prédominance sur les lois nationales de ces pays et pourra être invoquée directement contre des dispositions contraires de ces lois ou même en l'absence de dispositions. Une telle action sera exercée soit directement contre l'exploitant responsable, si les dispositions constitutionnelles relatives aux traités internationaux le permettent, soit sur la base de la responsabilité internationale de l'Etat contractant.¹² Dans le même ordre d'idée, la mise en vigueur des Conventions offrira une protection contre les modifications unilatérales du régime de réparation des dommages nucléaires qui seraient préjudiciables aux Parties contractantes.

19. La ratification des Conventions entraînera en outre un certain nombre de garanties qui peuvent être importantes pour les ressortissants des Etats

¹² Les Conventions de Paris (Annexe II) et de Vienne (art. XVIII) réservent expressément le maintien des droits découlant des règles de droit international public; la Convention de Bruxelles (art. XII.1) oblige les Etats contractants à prendre les mesures nécessaires à son application.

contractants qui sont victimes d'un dommage nucléaire dont la réparation est jugée par le tribunal d'un autre Etat contractant. Tout d'abord, il est stipulé que ces Conventions doivent être appliquées sans discrimination fondée sur la nationalité, le domicile ou la résidence (Paris, art. 14 a; Vienne, art. XIII; Bruxelles, art. XII.3; Complémentaire de Bruxelles, art. 8). Ensuite, en cas de défaillance des assureurs, l'Etat où se trouve l'installation responsable est obligé de payer lui-même les indemnités qui sont dues (Vienne, art. VII.1; pour la Convention de Paris, la même solution ressort de la rédaction de l'art. 10 selon lequel l'exploitant «devra être tenu... d'avoir et de maintenir» une assurance correspondant à sa responsabilité); une obligation similaire pèse sur l'Etat où est immatriculé le navire nucléaire responsable (art. III.2). Enfin, les Conventions prévoient que les indemnités et primes d'assurance, ainsi que les intérêts et dépens, sont librement convertibles dans les monnaies des Parties contractantes (Paris, art. 12; Vienne, art. XV; Bruxelles, art. XII.2).

20. Les dispositions les plus importantes pour les relations internationales sont celles qui fixent la compétence juridictionnelle et qui établissent le principe de l'unité de juridiction. D'après les Conventions de Paris (art. 13) et de Vienne (art. XI), c'est en règle générale le tribunal du lieu de l'accident qui est seul compétent, lorsque l'accident survient sur le territoire d'un Etat contractant; des dispositions précises déterminent la compétence juridictionnelle lorsque l'accident a lieu, en tout ou en partie, sur le territoire d'un autre Etat ou lorsqu'il y a incertitude sur le lieu de l'accident. La Convention de Bruxelles (art. X) rend compétents, au choix du demandeur, les tribunaux de l'Etat du pavillon du navire nucléaire en cause, ou ceux de l'Etat où le dommage a été subi.

Corollairement au principe de l'unité de juridiction, ces Conventions garantissent l'effet exécutoire, dans les pays contractants, des jugements définitifs rendus par le tribunal compétent après l'accomplissement des formalités d'exequatur et elles excluent tout nouvel examen au fond (Paris, art. 13 d; Vienne, art. XII; Bruxelles, art. XI.4).

Enfin, les Conventions nucléaires contiennent des dispositions sur le règlement des différends entre Parties contractantes relatifs à leur interprétation et à leur application.¹³

¹³ La Convention de Paris (art. 17) et la Convention complémentaire de Bruxelles (art. 17) donnent compétence pour ces litiges au Tribunal européen pour l'énergie nucléaire créé dans le cadre de l'ENEA par une Convention du 20 décembre 1957. La Convention de Vienne est accompagnée d'un Protocole facultatif sur le règlement obligatoire des différends par la Cour internationale de Justice. La Convention de Bruxelles sur les navires nucléaires prévoit un règlement par voie d'arbitrage et un recours à la Cour internationale de Justice (art. XX).

ANNEXE

CONVENTIONS INTERNATIONALES RELATIVES A LA RESPONSABILITE
EN MATIERE DE DOMMAGES NUCLEAIRES

ETAT DES SIGNATURES, RATIFICATIONS ET ADHESIONS

1. CONVENTION DE VIENNE RELATIVE A LA RESPONSABILITE
CIVILE EN MATIERE DE DOMMAGES NUCLEAIRES

La Convention a été élaborée par une conférence diplomatique mondiale que l'Agence internationale de l'énergie atomique avait organisée à Vienne, en mai 1963. La Conférence a aussi adopté un Protocole de signature facultative concernant le règlement obligatoire des différends.

a) Signatures

i) Jusqu'à présent, la Convention a été signée par les pays suivants¹⁴:

Chine	21 mai 1963
Colombie	21 mai 1963
Philippines	21 mai 1963
Yougoslavie	21 mai 1963
Espagne	6 septembre 1963
Royaume-Uni	11 novembre 1964
Cuba	10 décembre 1964
République Arabe Unie	19 août 1965
Argentine	10 octobre 1966

ii) Le Protocole de signature facultative a été signé par les pays suivants¹⁵:

Chine	21 mai 1963
Colombie	21 mai 1963
Philippines	21 mai 1963
Yougoslavie	21 mai 1963
Royaume-Uni	11 novembre 1964

b) Ratifications

i) Jusqu'à présent, la Convention a été ratifiée par les pays suivants:

Cuba	25 octobre 1965
République Arabe Unie	5 novembre 1965
Philippines	15 novembre 1965
Argentine	25 avril 1967

ii) Le Protocole de signature facultative a été ratifié par les Philippines le 15 novembre 1965.

¹⁴ Le Liban a paraphé la Convention le 21 mai 1963.

¹⁵ Le Liban a paraphé le Protocole de signature facultative le 21 mai 1963.

c) Adhésions

Les pays ci-après ont adhéré à la Convention:

Cameroun	6 mars 1964
Trinité-et-Tobago	31 janvier 1966
Bolivie	10 avril 1968

d) Entrée en vigueur

- i) Aux termes de l'article XXIII, la Convention entrera en vigueur trois mois après le dépôt du cinquième instrument de ratification et, pour tout Etat qui la ratifiera par la suite, trois mois après le dépôt de l'instrument de ratification de cet Etat.
 - ii) Aux termes de son article VII, le Protocole de signature facultative entrera en vigueur le même jour que la Convention ou le trentième jour suivant la date du dépôt du second instrument de ratification du Protocole ou d'adhésion à ce Protocole, si la deuxième date est postérieure.
2. CONVENTION SUR LA RESPONSABILITE CIVILE DANS LE DOMAINE DE L'ENERGIE NUCLEAIRE

a) Signatures

- i) La Convention a été signée à Paris, le 29 juillet 1960, par les pays européens ci-après:

Allemagne	Grèce	Portugal
Autriche	Italie	Royaume-Uni
Belgique	Luxembourg	Suède
Danemark	Norvège	Suisse
Espagne	Pays-Bas	Turquie
France		

- ii) La Convention de Paris a été modifiée par un Protocole additionnel qui est destiné à éviter un conflit éventuel avec la Convention de Vienne et a été signé à Paris, le 26 juillet 1964, par les Parties contractantes à la Convention de Paris.

b) Ratifications

- i) A l'heure actuelle, des instruments de ratification ont été déposés par les pays suivants:

Turquie	10 octobre 1961
Espagne	31 octobre 1961
Royaume-Uni	23 février 1966
France	9 mars 1966
Belgique	3 août 1966
Suède	1 ^{er} avril 1968
Grèce	12 mai 1970

- ii) Le Protocole additionnel à la Convention de Paris a été ratifié par les mêmes pays aux dates suivantes:

Espagne	30 avril 1965
Royaume-Uni	23 février 1966
France	9 mars 1966
Belgique	3 août 1966
Suède	1 ^{er} avril 1968
Turquie	5 avril 1968
Grèce	12 mai 1970

c) Adhésions

Il n'y a pas eu d'adhésions jusqu'à présent. Aux termes de l'alinéa c) de l'article II du Protocole additionnel, les adhésions à la Convention ne seront acceptées que si elles sont accompagnées d'une adhésion au Protocole additionnel.

d) Entrée en vigueur

Aux termes de l'alinéa b) de l'article 19, la Convention entrera en vigueur dès que cinq au moins des signataires auront déposé leur instrument de ratification. L'alinéa d) de l'article II du Protocole additionnel stipule que, pour le calcul du nombre de ratifications, il ne sera tenu compte que des signataires qui auront ratifié la Convention et ratifié ou confirmé le Protocole additionnel. En conséquence, la Convention est entrée en vigueur le 1^{er} avril 1968.

3. CONVENTION COMPLEMENTAIRE A LA CONVENTION DE PARIS
DU 29 JUILLET 1960 SUR LA RESPONSABILITE CIVILE DANS LE
DOMAINE DE L'ENERGIE NUCLEAIRE

a) Signatures

- i) La Convention complémentaire à la Convention de Paris a été signée à Bruxelles, le 31 janvier 1963, par les Parties contractantes à la Convention de Paris ci-après:

Allemagne	France	Pays-Bas
Autriche	Italie	Royaume-Uni
Belgique	Luxembourg	Suède
Danemark	Norvège	Suisse
Espagne		

- ii) La Convention complémentaire de Bruxelles a été également modifiée par un Protocole additionnel qui est destiné à éviter un conflit éventuel avec la Convention de Vienne et a été signé à Paris, le 28 janvier 1964, par les signataires de la Convention complémentaire.

b) Ratifications

La Convention complémentaire et le Protocole additionnel à cette Convention ont été ratifiés par les pays suivants:

Royaume-Uni	24 mars 1966
France	30 mars 1966
Espagne	27 juillet 1966
Suède	3 avril 1968

c) Adhésions

Il n'y a pas eu d'adhésions jusqu'à présent. Aux termes de l'alinéa c) de l'article II du Protocole additionnel, les adhésions à la Convention complémentaire ne seront acceptées que si elles sont accompagnées d'une adhésion au Protocole additionnel.

d) Entrée en vigueur

Aux termes du paragraphe c) de l'article 20, la Convention entrera en vigueur trois mois après le dépôt du sixième instrument de ratification. L'alinéa d) de l'article II du Protocole additionnel stipule que, pour le calcul du nombre de ratifications, il ne sera tenu compte que des signataires qui auront ratifié la Convention complémentaire et ratifié ou confirmé le Protocole additionnel.

4. CONVENTION RELATIVE A LA RESPONSABILITE DES EXPLOITANTS DE NAVIRES NUCLEAIRES

La Convention a été adoptée le 25 mai 1962 à l'issue d'une conférence diplomatique de droit maritime qui s'est tenue à Bruxelles.

a) Signatures

La Convention a été signée par les pays ci-après à l'issue de la Conférence:

Belgique	Irlande	Philippines
Chine	Libéria	Portugal
Corée	Malaysia	République Arabe Unie
Inde	Monaco	Yougoslavie
Indonésie	Panama	

et par les Pays-Bas le 30 décembre 1968.

b) Ratifications

La Convention a été ratifiée par le Portugal le 31 juillet 1968.

c) Adhésions

L'état actuel des adhésions se présente comme suit:

Madagascar	13 juillet 1965
Congo, République démocratique du	17 juillet 1967

d) Entrée en vigueur

La Convention entrera en vigueur trois mois après la date du dépôt des instruments de ratification par deux Etats dont au moins un Etat dont émane la licence (article XXIV, paragraphe 1).

ASPECTS DE LA REGLEMENTATION FRANÇAISE DANS LE DOMAINE NUCLEAIRE

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Abstract—Résumé—Аннотация—Resumen

ASPECTS OF FRENCH REGULATIONS IN THE NUCLEAR FIELD.

The French ruling on nuclear plants and work under radiation brings up certain theoretical and practical problems. The paper examines three important aspects: building licences, health physics, and transport responsibility insurance. Where building licences are concerned, the procedure for the authorization and supervision of basic nuclear facilities covers permit applications, the establishment of records, the delivery of a permit by decree. The permit covers general protection measures and special regulations governing the result achieved, rather than the means employed, and gives the authorities the possibility of judging, before the plant is put into service, the steps taken by the user to guarantee its safety (safety report, report on the monitoring systems and the results of tests, working instructions, ministerial permission). In the area of health physics, a complete ruling governing work under radiation has been issued within the last few years. In the professional field, it fixes the general principles of worker protection against ionizing radiations, safety norms and supervision principles, with special rules for large nuclear establishments. Where the public is concerned, health physics is dealt with from several aspects: possession and use of artificial and natural radioelements, transport, atmospheric and water pollution, contamination of the food chain and intervention in the event of an accident. With respect to transport responsibility insurance, nuclear responsibility agreements have been unable to prevail over maritime agreements and insurers refuse to cover ship owners for maritime law responsibilities from which they are not exempt. The future of nuclear transports will be assured by the priority of nuclear law over maritime law. It is hoped that the joint work of maritime and nuclear experts will lead to this result. In addition, an arrangement of the Paris agreement should provide for better compensation to victims of a fuel transport accident in transit on national territory.

ASPECTS DE LA REGLEMENTATION FRANÇAISE DANS LE DOMAINE NUCLEAIRE.

La réglementation française concernant les installations nucléaires et le travail sous rayonnements pose quelques problèmes théoriques et pratiques. Le mémoire en examine trois aspects importants: les licences des installations, la protection radiologique et l'assurance des responsabilités en cours de transport. En ce qui concerne les licences des installations, la procédure d'autorisation et de contrôle des installations nucléaires de base porte sur les demandes d'autorisation, l'instruction du dossier, la délivrance d'une autorisation par décret. L'autorisation comporte des mesures générales de protection et des prescriptions particulières sous forme d'obligations de résultat et non d'obligations de moyens, et donne aux autorités la possibilité d'apprécier, avant la mise en service de l'installation, les moyens employés par l'exploitant pour assurer la sûreté (rapport de sûreté, rapport sur les dispositifs de contrôle et les résultats des essais, consignes d'exploitation, autorisation ministérielle). Dans le domaine de la protection radiologique, une réglementation complète du travail sous rayonnements a été promulguée au cours des dernières années. Dans le domaine professionnel, elle fixe des principes généraux de protection des travailleurs contre les rayonnements ionisants, les normes de sécurité, les principes de surveillance, avec des règles particulières pour les grands établissements nucléaires. En ce qui concerne le public, elle traite de la protection sanitaire sous différents aspects: détention et utilisation des radioéléments artificiels et naturels, transports, pollution atmosphérique et pollution des eaux, contamination de la chaîne alimentaire et intervention en cas d'accident. Pour ce qui est de l'assurance des responsabilités en cours de transport, les conventions de responsabilité nucléaire n'ont pu prévaloir sur les conventions maritimes, et les assureurs refusent aux armateurs la garantie des responsabilités de droit maritime dont ils ne sont pas déchargés. L'avenir des transports nucléaires est conditionné par la primauté du droit nucléaire sur le droit

maritime. Les travaux communs des experts maritimes et nucléaires permettent d'espérer que ce but pourra être atteint. D'autre part, un aménagement de la Convention de Paris devrait permettre une meilleure indemnisation des victimes d'un accident de transport de combustibles en transit sur le territoire national.

АСПЕКТЫ ЗАКОНОДАТЕЛЬСТВА ФРАНЦИИ В ОБЛАСТИ ИСПОЛЬЗОВАНИЯ ЯДЕРНОЙ ЭНЕРГИИ.

Инструктивные положения Франции по ядерным установкам и порядку работы в условиях радиации поднимают некоторые теоретические и практические вопросы. Когда вопрос касается выдачи лицензий на сооружение установок, процедура выдачи полномочий и надзора за основными ядерными устройствами предусматривает подачу заявлений на получение лицензий, ведение учетных документов, проведение лицензии законодательным актом. Лицензия включает в себя принятие общих мер по обеспечению защиты и специальные правила, определяющие скорее полученный результат, а не используемые для этого средства. Она дает возможность соответствующим органам до ввода установки в эксплуатацию оценить меры, предпринятые потребителем в целях обеспечения безопасности установки (отчет по безопасности, отчет по дозиметрическим системам и результатам испытаний, правила эксплуатации, разрешение министерства). По вопросам биологической защиты были опубликованы в течение нескольких последних лет инструктивные положения, регламентирующие работу в условиях радиации. В профессиональной области эти положения определяют общие принципы защиты персонала от ионизирующих излучений, нормы безопасности и принципы надзора, а также предусматривают специальные правила в отношении крупных ядерных центров. Поскольку этот вопрос затрагивает общественность, биологическая защита рассматривается в нескольких аспектах: наличие и использование искусственных и естественных радиоэлементов, перевозка, загрязнение атмосферы и воды, заражение пищевой цепочки и вмешательство в случае аварии. Что касается страхования и ответственности при перевозке ядерных веществ, то соглашения об ответственности за ядерный ущерб не могут превалировать над морскими соглашениями, и поэтому страховщики отказываются страховать юридическую ответственность владельцев судов за морские перевозки, от которой владельцы судов не освобождены. Будущее перевозок ядерных веществ обуславливается приоритетом ядерного права над морским. Ожидается, что совместная работа экспертов по морскому и ядерному праву приведет к такому результату. Кроме того, Заключение Парижского соглашения должно обеспечить несколько лучшую компенсацию тем, кто станет жертвой в случае аварии при перевозке топлива транзитом по национальной территории данного государства.

DIVERSOS ASPECTOS DE LA REGLAMENTACION FRANCESA EN MATERIA NUCLEAR.

La reglamentación francesa relativa a las instalaciones nucleares y a los trabajadores sometidos a las radiaciones plantea algunos problemas tanto de índole teórica como práctica. La memoria examina tres aspectos importantes: la autorización de las instalaciones, la protección radiológica y la aseguración de la responsabilidad durante el transporte. En lo que respecta a la autorización de las instalaciones, el procedimiento de autorización y control de las instalaciones nucleares comprende las solicitudes de autorización, la apertura de expedientes y la concesión de autorizaciones por decreto. La autorización en sí entraña medidas de protección de carácter general, así como normas específicas que prescriben un resultado y no los medios para alcanzarlo, y permite juzgar a las autoridades, antes de la puesta en servicio de la instalación, los medios utilizados por el explotador para garantizar la seguridad de ésta (estudio de seguridad, informe sobre los dispositivos de control y los resultados de los ensayos, normas de trabajo y autorización ministerial). En el campo de la protección radiológica, en estos últimos años se ha promulgado una reglamentación completa de las actividades laborales desarrolladas bajo las radiaciones. En la esfera profesional, esta reglamentación fija principios generales de protección de los trabajadores contra las radiaciones ionizantes, normas de seguridad y principios de vigilancia, junto con reglas especiales para las grandes instalaciones nucleares. Por lo que respecta al público, la reglamentación versa sobre distintos aspectos de protección de la salud: posesión y empleo de radioelementos artificiales y naturales, transporte, contaminación atmosférica y de las aguas, contaminación de la cadena alimentaria e intervención en caso de accidente. En cuanto a la aseguración de la responsabilidad durante el transporte, las convenciones en materia de responsabilidad nuclear no han podido prevalecer sobre las convenciones marítimas y los aseguradores rehúsan cubrir a los armadores respecto de las responsabilidades del derecho marítimo, de las que no están exentos. El porvenir del transporte nuclear dependerá de la primacía del derecho nuclear sobre el derecho marítimo. Los trabajos comunes de los expertos marítimos y nucleares inducen a creer que se podrá alcanzar esta meta. Además, una modificación del Convenio de París debería hacer posible una indemnización más equitativa a las víctimas de los accidentes que se produzcan en el transporte de combustibles nucleares en tránsito por el territorio nacional.

Le droit français comporte plusieurs réglementations ayant pour objet d'assurer la protection des populations au sens large contre les dangers présentés par les activités d'une industrie moderne. Ces réglementations tendent à atteindre deux objectifs: la protection des travailleurs et la protection de l'environnement. Dans le domaine nucléaire, deux techniques sont concurremment mises en œuvre pour atteindre ces deux objectifs. La « sûreté nucléaire » consiste à définir les dispositions à prendre dans la réalisation et l'exploitation de l'installation pour qu'un accident ne puisse s'y produire, ou, si un accident survient malgré les précautions prises, en limiter les conséquences. La « radioprotection » fixe tout d'abord certaines données qui servent de base à la sûreté nucléaire; elle consiste aussi dans la mise en œuvre de moyens de protection et de mesure, et dans une surveillance médicale spéciale du personnel.

Les procédures existant dans le droit français n'étaient pas adaptées aux problèmes nouveaux posés par l'utilisation de l'énergie atomique. Il a donc été nécessaire d'élaborer une réglementation spéciale.

Ce besoin d'une réglementation particulière s'est aussi fait sentir dans le domaine de la responsabilité civile et de son assurance.

1. LICENCES DES INSTALLATIONS

La sûreté des installations nucléaires fait l'objet d'une procédure d'autorisation et de contrôle instituée par le décret n° 1 228 du 11 décembre 1963. Cette procédure concerne les installations les plus importantes; les autres installations sont réglementées dans le cadre de la loi du 19 décembre 1917 relative aux installations dangereuses, incommodes et insalubres dont il ne sera pas question dans cet exposé.

La procédure prévue par le décret de 1963 est assez souple pour qu'elle puisse s'adapter aux diverses situations qui peuvent se présenter. Elle permet notamment de fixer dans le décret d'autorisation les prescriptions auxquelles l'exploitant devra se conformer.

1.1. La procédure

Seuls les exploitants de certaines installations nucléaires sont tenus de se conformer aux dispositions du décret du 11 décembre 1963.

1.1.1. Les installations concernées

La liste des installations soumises au décret a été établie à partir de la liste des installations soumises aux Conventions de Paris et de Vienne sur la responsabilité civile dans le domaine nucléaire.

Cette liste comprend les réacteurs et leurs installations annexes; les usines de préparation, de fabrication ou de transformation de substances radioactives d'une certaine importance; les usines de séparation des isotopes de l'uranium ou du plutonium; les usines de traitement d'uranium, de plutonium ou de thorium irradiés; les usines de traitement de déchets radioactifs, des installations destinées au stockage, au dépôt ou à l'utilisation de quantités de substances radioactives supérieures à certains seuils fixés par arrêté ministériel. Il s'y ajoute les accélérateurs dont la puissance est supérieure à 300 MeV qui ne figurent pas sur la liste des con-

ventions. Il faut aussi remarquer que les usines traitant de l'uranium naturel en quantité relativement importante sont aussi soumises à la procédure du décret de 1963, alors qu'elles ne sont pas concernées par le régime spécial de responsabilité des Conventions de Paris et de Vienne.

1.1.2. La demande d'autorisation et l'instruction du dossier

L'exploitant adresse une demande d'autorisation de création au ministre chargé de l'énergie atomique, c'est-à-dire au ministre du développement industriel et scientifique. Lorsque l'organisme ou l'entreprise relève du contrôle d'un autre ministre, la demande lui est également adressée. A ce stade de la procédure, le ministre de la santé publique, le ministre de l'équipement et le ministre de l'intérieur sont informés de la demande et peuvent de ce fait adresser leurs observations au ministre du développement industriel et scientifique.

L'exploitant joint à sa demande un dossier comprenant tous les renseignements connus au moment du dépôt de la demande concernant l'installation. Ces renseignements sont plus ou moins complets et définitifs selon la nature de l'installation. Ils portent donc sur le site, sur la nature et sur les caractéristiques essentielles de l'installation.

Le dossier d'une installation nucléaire de base fait l'objet d'un examen technique par les services et commissions du Commissariat à l'énergie atomique spécialisés en matière de sûreté. En ce qui concerne les centrales nucléaires, un groupe ad hoc constitué pour chaque Centrale associe aux travaux d'étude du dossier les représentants de la direction de l'électricité au ministère du développement industriel et scientifique et les représentants d'Electricité de France, ainsi que les inspecteurs chargés du contrôle.

La demande d'autorisation de création d'une installation nucléaire de base est en principe soumise à une enquête publique, conformément au principe bien établi du droit français qui exige que tout intéressé soit mis en mesure de faire connaître ses observations ou objections à un projet susceptible de le concerner (par exemple, enquête préalable à la déclaration d'utilité publique en matière d'expropriation, enquête de commodo et incommodo prévue par la loi de 1917 précitée).

Toutefois, cette enquête publique, dont la procédure a été définie par un texte particulier, n'a pas lieu d'être ouverte s'il a été procédé à une enquête préalable à la déclaration d'utilité publique ou s'il s'agit d'implanter une installation nouvelle sur un site où existe déjà une autre installation nucléaire ayant fait l'objet d'une enquête. Les intéressés ont eu en effet dans ces hypothèses la possibilité de faire connaître leurs observations.

En fait, jusqu'à présent, seules des enquêtes préalables à la déclaration d'utilité publique ont été ouvertes, cette déclaration présentant en effet l'avantage de produire des effets multiples.

Parallèlement à l'enquête publique, les observations sur le projet des diverses administrations ou collectivités départementales sont recueillies.

Toutes les observations recueillies au cours de l'enquête sont transmises au ministre chargé de l'énergie atomique.

L'examen technique du dossier permet de définir un certain nombre de prescriptions techniques particulières qui seront imposées à l'exploitant. Ces prescriptions sont introduites dans un projet de décret d'autorisation.

Celui-ci contient d'autres dispositions plus générales qui permettent à l'autorité publique de veiller à ce que la sûreté de l'installation soit assurée au stade de la construction, lors de la mise en service et au cours de l'exploitation de l'installation. Ces dispositions générales seront exposées plus loin.

Le projet de décret est examiné par une commission interministérielle qui réunit des représentants des ministères et organismes concernés, soit par l'étude ou la promotion de l'énergie nucléaire, soit par la protection contre ses effets ou l'intervention en cas d'accident.

Le décret d'autorisation est signé du Premier ministre, du ministre de l'énergie atomique et, le cas échéant, du ministre dont relève l'installation, sur avis conforme du ministre de la santé publique.

1.2. Les prescriptions auxquelles l'exploitant est soumis

Le décret du 11 décembre 1963 prévoit dans son article 4 que l'autorisation fixe les mesures auxquelles devra se conformer l'exploitant. Comme on l'a vu, la procédure d'autorisation est une procédure solennelle par décret pris au moment de la création de l'installation. Toutes les options techniques ne sont pas définitivement arrêtées à ce moment-là. Aussi est-il évident que l'autorisation de création doit permettre à l'autorité publique de vérifier postérieurement que toutes les mesures nécessaires pour assurer la sûreté de l'installation soient prises.

Les prescriptions imposées à l'exploitant par le décret d'autorisation sont pour la plupart des obligations de résultat, c'est-à-dire qu'elles ne lui imposent pas les moyens à utiliser pour obtenir la sûreté. Cependant, l'autorité publique peut dès l'autorisation de création, d'après les éléments du dossier à ce moment-là, imposer certaines prescriptions techniques qui paraissent essentielles pour assurer la sûreté et s'analysent alors en obligations de moyens; elle se réserve aussi dans le décret le pouvoir d'approuver les principaux moyens matériels et d'organisation qui doivent être mis en œuvre avant les essais, avant la mise en exploitation et tout au long de l'exploitation de l'installation.

Ainsi, avant de procéder aux essais, l'exploitant doit adresser au ministre:

- un rapport de sûreté, c'est-à-dire une description de l'installation présentée sous l'angle de la sûreté,
- un rapport sur les opérations de démarrage.

Avant la mise en exploitation, il soumet au ministre:

a) un rapport

- sur les dispositifs de contrôle installés pour assurer le respect des prescriptions particulières qui lui sont imposées;
- sur les épreuves effectuées;
- sur les résultats de montée en puissance (réacteurs);

b) les consignes générales d'exploitation.

L'installation fait alors l'objet d'une autorisation de mise en exploitation par le ministre, ou bien les documents ci-dessus font l'objet d'une approbation ministérielle (centrales nucléaires). De cette manière, les

moyens mis en œuvre par l'exploitant font l'objet d'une approbation de l'autorité publique.

Toute modification aux documents ci-dessus et particulièrement aux consignes générales d'exploitation doit être communiquée au ministre chargé de l'énergie atomique. Toute modification importante à l'installation qui affecterait les conditions de la sûreté ou tout changement d'exploitant doivent faire l'objet d'une autorisation accordée par décret.

Enfin, l'autorisation prévoit que les autres réglementations de sûreté continuent à s'appliquer. Par exemple: celles relatives à l'utilisation du courant électrique ou des appareils à pression de gaz ou de vapeur, réglementation de l'urbanisme. Enfin, le ministre notifie à l'exploitant des prescriptions d'effet équivalent à celles qui auraient été prises si la loi de 1917 s'appliquait dans les installations nucléaires.

Le contrôle des installations nucléaires est assuré par des inspecteurs des établissements classés (établissements soumis à la loi du 19 décembre 1917) spécialisés en matière de sûreté. Ces inspecteurs peuvent se faire assister d'experts nucléaires. La surveillance de l'environnement à l'égard de la pollution radioactive est assurée par le Service central de protection contre les rayonnements ionisants, qui dépend du ministère de la santé publique.

En conclusion, il convient d'attirer l'attention sur le caractère résolument pragmatique de la réglementation française qui a su unir une procédure assez solennelle d'autorisation accordée par décret à l'établissement de prescriptions techniques adaptées à chaque type d'installation et à l'avancement de la science et de la technologie dans ce domaine. Cette réglementation pourra dans l'avenir faire l'objet d'améliorations.

2. PROTECTION RADIOLOGIQUE

Au cours des dernières années, la mise en place des moyens de protection du travailleur et des personnes du public s'est accompagnée d'une réglementation très complète. Elle tient compte à la fois des impératifs sanitaires, des disponibilités techniques mais aussi des réalités économiques.

Une circulaire du ministre de la santé publique du 23 février 1957 relative à la protection contre les radiations ionisantes rappelait les dispositions fragmentaires existant alors dans la réglementation française pour la protection. Elle soulignait que l'expérience insuffisante du risque est malaisément conciliable avec l'exigence de précision et de rigueur d'une réglementation, et que les dispositions nouvelles devraient faire l'objet d'un constant effort d'adaptation.

Une circulaire du 3 juin 1957 édictait un certain nombre de recommandations générales visant la protection contre les radiations ionisantes et introduisait pour la première fois en France la notion de dose maximale admissible.

A la suite de la mise au point des normes de base par EURATOM celles-ci ont été mises en application en France par une circulaire du 7 mars 1962.

Le décret du 20 juin 1966 devait fixer les principes généraux de protection tant pour les travailleurs que pour la population. Ce texte constitue actuellement la base de la réglementation française en matière de protection radiologique.

2.1. Domaine professionnel

Il existait en France un certain nombre de textes réglementaires applicables aux travaux sous rayonnements, en particulier le décret du 5 décembre 1934 relatif aux précautions à prendre dans l'emploi et la manipulation des corps radioactifs et la circulaire du 3 juin 1957 susvisée.

Le décret du 15 mars 1967, directement inspiré des principes de protection du décret du 20 juin 1966, contient un certain nombre de dispositions administratives, techniques et médicales visant à assurer la protection des travailleurs. Des arrêtés en date du 18 au 24 avril 1968 ont été pris en application de ce décret.

Cependant, ces textes conçus pour les petits utilisateurs de radioéléments ne s'appliquent pas aux installations nucléaires de base définies par le décret du 11 décembre 1963.

A juste titre, les ministères intéressés envisagent donc la préparation d'un texte mieux adapté à la protection des travailleurs dans les grandes installations. Il faut d'ailleurs noter que ces grandes installations sont dotées depuis de nombreuses années de services spécialisés et qu'elles ont établi pour elles-mêmes des procédés de surveillance draconiens. Des problèmes peuvent toutefois se soulever, en particulier dans la définition d'installations relevant, dans un même établissement nucléaire, d'une réglementation du décret de 1963 ou d'une réglementation de la loi de 1917.

Les principes essentiels des décrets de 1966 et de 1967 et des arrêtés de 1968 peuvent être groupés en un certain nombre de rubriques relatives aux équivalences de dose, à la surveillance et au respect des limites de sécurité et d'action.

a) Les données relatives aux équivalences de dose portent référence des recommandations de la CIPR et des directives d'EURATOM. Elles concernent, en conditions normales de travail, les personnes directement affectées ou professionnellement exposées à des travaux sous rayonnements se trouvant dans une zone contrôlée et où tous les moyens de protection physiques et médicaux sont mis en œuvre. Les valeurs fixées sont les valeurs limites de l'irradiation (quel que soit le mode d'irradiation dans les organes) dans les différentes parties du corps humain. Elles sont en termes de doses prévues pour trois mois consécutifs sans dépasser 3 rems. Il faut noter que les hommes peuvent subir 3 à 12 rems en irradiation externe et d'une façon exceptionnelle et concertée à condition que cela ne se reproduise pas; par contre, les femmes et principalement les femmes enceintes ne doivent pas dépasser 1,5 rem durant leur année de grossesse, limite que l'on retrouve pour les personnes non directement affectées à des travaux sous rayonnements (l'équivalence de dose reçue au cours d'une année ne doit pas dépasser 1,5 rem/an). Il se pose aussi un problème d'âge, ces normes sont prévues pour des travailleurs ayant atteint 18 ans, âge légal du travail. On considère qu'en dessous de 18 ans (entre 16 et 18 ans) la dose ne doit pas dépasser 5 rem/an qui correspond à l'équivalence de dose maximale admissible pour la population dans son ensemble. Enfin, lors d'une irradiation exceptionnelle non concertée, la dose résultante est considérée comme n'entraînant pas de risque sérieux si elle est inférieure à 25 rems.

b) Les décrets du 20 juin 1966 et du 15 mars 1967 prévoient « qu'il appartient à l'exploitant de l'établissement d'assurer la protection contre les rayonnements ionisants des personnes travaillant à l'intérieur de l'établissement ainsi que de celles qui sont amenées à y pénétrer à quelque titre que ce soit ». Il est précisé « qu'une zone contrôlée doit être établie partout où les conditions d'expositions sont telles que les personnes sont susceptibles de recevoir des équivalences de dose supérieures à 1,5 rem ».

En fait, la spécification et la signalisation des zones contrôlées, l'examen et le contrôle des dispositifs de protection, l'évaluation des irradiations et contaminations constituent l'ensemble des mesures de protection collective et individuelle, lesquelles contribuent à la sécurité des installations et à l'hygiène du travail, ces mesures étant suivies par des personnes qualifiées.

c) A ces différents contrôles s'ajoute la surveillance médicale qui est définie dans les textes précités, c'est-à-dire: « aucune personne ne doit être affectée directement à des travaux sous rayonnements sans une attestation médicale prouvant son aptitude à de tels travaux ». La surveillance médicale comprend: un examen d'embauche et des examens périodiques dont le rythme n'est pas fixe, ils dépendront de l'état de santé de l'individu et des conditions de travail, toutefois, il est stipulé que l'écart ne peut être supérieur à un an.

En outre, en cas d'irradiation externe importante comme en cas de contamination du travailleur, des examens médicaux spéciaux seront effectués. Un dossier médical pour chaque travailleur doit être tenu à jour. Le rôle du médecin défini particulièrement par l'arrêté de 1968 est important, car il a la responsabilité, vis-à-vis de l'exploitant, de désigner les postes de travail et de maintenir ou non les travailleurs à ces postes. Il se base pour cela sur un certain nombre de critères (antécédents, examens cliniques, hématologiques, etc.).

Les textes étendent l'obligation de conserver les procès-verbaux se rapportant à l'évaluation des doses individuelles à toute la durée de la vie de l'intéressé et en tout cas pendant les trente années qui suivent l'arrêt de travail sous rayonnements. La conservation des renseignements relatifs à l'efficacité des moyens de contrôle physique et à l'évaluation des doses individuelles permet aux experts qualifiés de se faire une idée plus précise de l'importance du risque. En outre, cette documentation aide le médecin agréé dans une interprétation plus objective des examens auxquels il procède.

Les décrets de 1966 et 1967 sont complétés par un certain nombre d'arrêtés du 18 au 24 avril 1968 qui en précisent les modalités d'application. Cinq autres arrêtés concernent respectivement:

- les modalités d'agrément des organismes habilités à effectuer les contrôles prévus par le décret;
- la périodicité des contrôles à effectuer sur les appareils générateurs électriques de rayonnements ionisants et les sources scellées ainsi que sur leurs dispositifs de protection;
- les conditions d'utilisation des dosimètres individuels destinés au contrôle des équivalents de dose reçus par les travailleurs directement affectés à des travaux sous rayonnements et exposés au risque d'irradiation externe;

- l'approbation des méthodes de contrôle élaborées par le ministère de la santé;
- des recommandations aux médecins exerçant la surveillance médicale prévue par le décret.

Le sixième arrêté précise le régime particulier applicable aux installations du Commissariat à l'énergie atomique en raison de la compétence de cet organisme.

2.2. Domaine public

Le décret du 20 juin 1966 fixe les équivalents de dose admissible pour les personnes du public et la population dans son ensemble qui comprend les personnes du public et les travailleurs. Il oblige l'exploitant à prendre toutes mesures pour que les personnes du public se trouvant à l'intérieur de l'établissement ne reçoivent pas du fait de l'activité de l'établissement des équivalents de dose supérieurs à ceux fixés.

L'exploitant doit surveiller les rejets de son installation.

D'autres textes se préoccupent de la protection contre les rayonnements ionisants; ils sont à la fois moins complets que les précédents et plus longs, car ils concernent aussi bien le travailleur que le public. Ils traitent de:

- l'utilisation et la détention des radioéléments artificiels;
- l'utilisation des radioéléments naturels;
- les transports;
- la pollution atmosphérique;
- la pollution des eaux;
- la contamination des produits alimentaires d'origines animale et végétale;
- l'intervention en cas d'accidents.

1° En ce qui concerne le commerce et l'utilisation des radioéléments artificiels, on sait qu'en France ils sont étroitement réglementés. Cette législation spéciale relative aux radioéléments artificiels qui réside dans le Code de la santé publique (art. L 31 et suivants et A 5230 et suivants) prévoit que la préparation, l'utilisation, l'importation, l'exportation de radioéléments artificiels sous quelque forme que ce soit ne peuvent être effectuées que par le Commissariat à l'énergie atomique ou les personnes physiques ou morales spécialement autorisées à cet effet et après avis d'une commission interministérielle.

2° Quant à l'utilisation des radioéléments naturels, elle n'est pas encore strictement organisée. Des textes sont à l'étude mais leur sortie présente quelques difficultés. En bref, on peut citer l'art. 44.2 du Code de la santé qui précise que les radiations ionisantes ne peuvent être utilisées sur le corps humain qu'à des fins exclusivement médicales et thérapeutiques. On peut ajouter aussi l'interdiction d'introduire des substances radioactives dans les jouets et les aliments, et la réglementation sur les substances vénéneuses.

3° La réglementation sur les transports dérive directement des textes internationaux (ce qui se comprend). On peut dire cependant que le transport des matières radioactives qui se développe sans cesse par l'utilisation

croissante de ces matières comporte des risques et pose des problèmes techniques principalement d'emballage; les conditions de sûreté reposent donc surtout sur l'emballage. Les textes officiels en vigueur sont récents et portent sur les transports par terre, air et mer. En bref, le ministre des transports, après avoir agréé les modèles d'emballage, donne son approbation dans le cas d'expéditions des groupes I et II, mais dans le cas d'expéditions du groupe III une autorisation spéciale du ministre est nécessaire pour chaque expédition.

4° En ce qui concerne la pollution du milieu naturel, il est bien évident que tous les départements ministériels sont intéressés par les mesures de protection. Deux textes assez récents ont précisé certains points, il s'agit:

- a) de la loi du 2 août 1961 relative à la lutte contre la pollution atmosphérique, les odeurs et le bruit qui a prévu en son article 2 que des décrets détermineront « les cas et conditions dans lesquels pourra être interdite ou réglementée l'émission dans l'atmosphère de fumées, suies, poussières ou gaz toxiques corrosifs, odorants ou radioactifs »; ce texte a fixé aussi que les agents du ministère de la santé seraient chargés de la surveillance, à l'extérieur des installations;
- b) de la loi du 16 décembre 1964 sur la pollution des eaux qui précise dans son article 2 en ce qui concerne la mer: « est interdit le déversement ou l'immersion dans les eaux de la mer de matières de toute nature, en particulier de déchets industriels et atomiques, susceptibles de porter atteinte à la santé publique ainsi qu'à la faune et à la flore sous-marines et de mettre en cause le développement économique et touristique des régions côtières »; en ce qui concerne les déversements existants, le préfet déterminera le délai dans lequel la présente interdiction leur est applicable.

Toutefois, le préfet pourra, après enquête publique, autoriser de réglementer le déversement ou l'immersion visés à l'alinéa ci-dessus dans le cas où ceux-ci pourront être effectués dans des conditions telles qu'elles garantissent l'innocuité et l'absence de nuisance du déversement ou de l'immersion.

Mais cette loi a une portée beaucoup plus large puisqu'elle conduit à une véritable politique de l'eau en France. Pour information, il est intéressant de signaler que cette réglementation pose le principe de l'indispensable corrélation entre les problèmes d'alimentation en eaux potables, industrielles ou agricoles et ceux d'évacuation des eaux usées, en jumelant pour les deux la gestion.

En ce qui concerne les grands établissements nucléaires, par l'intermédiaire de protocoles avec le ministre de la santé, ils ont établi un système qui conduit ce ministère à effectuer une surveillance propre au ministère.

5° Enfin, pour terminer ce survol du domaine public, on peut citer dans le cas d'accidents l'intervention du service de la Protection civile du ministère de l'intérieur. Il intervient là dans le cadre d'une instruction interministérielle sur l'établissement de plans de défense contre l'irradiation et la contamination radioactive accidentelle.

Dans ce même cadre, il est nécessaire de citer l'arrêté faisant mention du Comité national d'experts médicaux qui intervient en cas de sinistres dus à la contamination radioactive et qui assiste le ministre de l'intérieur et les préfets.

3. ASSURANCE DES RESPONSABILITES EN COURS DE TRANSPORT DE SUBSTANCES NUCLEAIRES

L'expérience vécue par les exploitants d'installations nucléaires européennes pendant ces dix dernières années a mis l'accent sur deux sortes de difficultés qui viennent entraver les transports internationaux de substances nucléaires. La première, en date et en importance, est du domaine des transports maritimes. Elle est due essentiellement aux particularités du droit maritime en matière de responsabilité, ce qui explique que les experts maritimes et nucléaires aient dû joindre leurs efforts au sein de leurs organismes les plus représentatifs, l'Agence internationale de l'énergie atomique, l'Agence européenne pour l'énergie nucléaire de l'OCDE, l'Organisation intergouvernementale consultative de la navigation maritime et le Comité maritime international, pour rechercher une solution qui permette de sortir de l'impasse actuelle et donne satisfaction tout à la fois aux exploitants d'installations nucléaires et aux propriétaires de navires.

Depuis le Colloque de Monaco d'octobre 1968 sur la responsabilité civile et l'assurance en matière de transports maritimes de substances nucléaires, il ne fait plus de doute que l'avenir de ces transports est tout entier tributaire de l'aboutissement de ces travaux communs et de leur couronnement par l'adoption, la signature et la ratification la plus large possible d'une nouvelle Convention maritime sur la responsabilité en matière de transports de substances nucléaires.

Seule une telle Convention est en effet en mesure de reconnaître la primauté du droit nucléaire sur le droit maritime dans ce domaine, principe nouveau, diamétralement opposé à la règle juridique actuelle. Refuser de partager cette conviction équivaudrait à consentir à la paralysie définitive du commerce nucléaire maritime, et par voie de conséquence à la paralysie de tout un secteur d'activité nucléaire de première importance.

Force est de constater tout d'abord que l'espoir des auteurs de la Convention de Paris du 29 juillet 1960 et de la Convention de Vienne du 21 mai 1963 d'avoir trouvé la meilleure solution dans le respect des règles du droit maritime s'est rapidement trouvé déçu dans la pratique. Les uns et les autres s'étaient cependant efforcés de ne modifier en rien le régime de responsabilité des transporteurs maritimes, les deux textes reconnaissant expressément être sans effet sur l'application des accords internationaux de transport, en vigueur ou ouverts à la signature, à la ratification ou à l'adhésion à la date des Conventions nucléaires. Bien plus, les transporteurs tenus de réparer les conséquences d'un accident nucléaire en vertu de l'un de ces accords internationaux – ou même de la législation d'un Etat non contractant – se voyaient subrogés dans les droits dont disposaient les victimes contre l'exploitant de l'installation, seul et unique responsable au titre des Conventions nucléaires.

Cette construction juridique fut totalement ruinée par la décision des « Protecting Clubs » de retirer toute garantie à leurs adhérents, pour les transports portant sur des matières fissiles, des combustibles nucléaires ou déchets irradiés, aucun armateur ne voulant se risquer à conserver à sa charge, sans le secours de sa mutuelle professionnelle, les conséquences financières d'une responsabilité susceptible d'être engagée au-delà des limites conventionnelles ou contractuelles, quelquefois même de façon illi-

mitée conformément à une jurisprudence ayant donné lieu à quelques arrêts français bien connus.¹

De ce refus découle l'impossibilité pour un exploitant de trouver un armateur qui accepte ce genre de substances, dont la plupart, ne serait-ce que les combustibles irradiés à renvoyer dans les usines de retraitement, ne peuvent, en raison du poids et de l'encombrement des châteaux de protection, être transportés autrement que par voie maritime.

Puisque les transporteurs maritimes se trouvent privés de la couverture de leurs mutuelles professionnelles, et qu'ils s'estiment insuffisamment protégés par les droits de recours qui leur sont offerts par le droit nucléaire, recours qui n'empêchent pas il est vrai que leur responsabilité soit d'abord mise en jeu, et qui sont forcément plafonnés au montant maximal de responsabilité des exploitants nucléaires, il n'est pas d'autre voie pour les inciter à accepter les transports de substances nucléaires que de modifier le droit de telle façon qu'ils n'encourent plus aucune responsabilité en cas d'accident nucléaire. Et rien ne s'oppose à ce qu'ils soient ainsi déchargés de toute responsabilité dès que la réparation des dommages causés est acquise aux victimes au titre de la responsabilité de l'exploitant de l'installation concernée, en application des Conventions nucléaires.

Pour parvenir à cette fin, on pouvait être tenté, puisque tout le mal semblait venir de là, de modifier les Conventions nucléaires de Paris et de Vienne, soit en stipulant expressément qu'elles l'emportaient sur les Conventions maritimes pour ce qui est du régime de responsabilité, soit en supprimant purement et simplement la disposition en vertu de laquelle il n'était pas porté atteinte aux règles du droit maritime.

Mais ces solutions s'avérèrent rapidement illusoires. Comment imaginer en effet, les signataires des Conventions maritimes étant beaucoup plus nombreux que les signataires des Conventions nucléaires (16 pour la Convention de Paris, 9 pour la Convention de Vienne), qu'une disposition de ces dernières puisse être opposable à l'ensemble des signataires des Conventions maritimes — et cela bien qu'elle soit favorable à l'armateur tenu pour responsable de l'accident aux termes du droit maritime.

A plus forte raison, si l'on choisissait de supprimer toute disposition à cet égard dans les Conventions nucléaires, le seul résultat obtenu, qui ne résolvait rien, aurait été que l'Etat signataire à la fois d'une Convention maritime et d'une Convention nucléaire, se serait trouvé avoir pris des engagements contradictoires, la Convention nucléaire canalisant la responsabilité sur une personne unique, l'exploitant d'une installation, et les Conventions maritimes permettant toujours de rechercher l'armateur ou le propriétaire du navire.

En vérité, la voie était toute tracée par la Convention internationale relative à la responsabilité des exploitants de navires nucléaires du 25 mai 1962, ou par le protocole de 1968 à la Convention de 1924 en matière de connaissance, qui font porter toute la responsabilité des conséquences d'un accident nucléaire sur l'exploitant de l'installation en cause, à l'exclusion de tout autre responsable, à quelque titre que ce soit.

C'est au droit maritime de reconnaître la primauté du droit nucléaire, et ce dernier, inchangé, stipulant toujours qu'il n'affecte pas les règles

¹ Arrêts Lamoricière, Cour de Cassation, Chambre commerciale des 19 juin 1951 et 23 janvier 1952; Arrêts Champollion, Cour de Paris du 4 juillet 1956 et Cour de Cassation, Chambre civile du 23 janvier 1959.

du droit maritime en matière de responsabilité, il est sûr que les armateurs ne risquent plus de se voir recherchés sur la base des conventions maritimes ainsi amendées.

Il ne restait plus qu'à trancher le point de savoir s'il était préférable de modifier séparément toutes les Conventions maritimes susceptibles de trouver application à l'occasion d'un transport de substances nucléaires, ou de promulguer une nouvelle Convention indépendante amendant toutes les Conventions existantes. Cette dernière solution est manifestement plus simple et plus sûre.

Plus simple car elle ne nécessite qu'un seul instrument et non pas autant d'instruments que de Conventions. Plus sûre, car elle n'exige pas de recensement des Conventions à modifier, opération d'autant plus délicate que toutes les Conventions maritimes ne comportent pas à proprement parler de clauses de responsabilité, et que certaines contiennent cependant des dispositions limitant les responsabilités encourues suivant les principes généraux du droit maritime.

Le projet de Convention maritime élaboré de concert par les experts nucléaires et maritimes de l'AIEA, de l'ENEA, de l'IMCO et du CMI doit trouver un accueil favorable auprès de tout Etat soucieux à la fois du développement de son industrie nucléaire et de l'avenir de son commerce nucléaire par voie maritime. Armateurs et exploitants nucléaires ont le même intérêt à inciter leurs gouvernements à signer cette Convention dont l'objectif ne sera véritablement réalisé que si elle est ratifiée par le plus grand nombre de pays signataires des Conventions maritimes, qu'ils soient ou non parties aux Conventions nucléaires de Paris ou de Vienne.

La seconde entrave au transport des substances nucléaires, à laquelle il a été fait allusion dans notre préambule, risque de se produire, cette fois-ci dans le domaine des transports terrestres, quand un exploitant d'installation nucléaire veut faire transiter un envoi de substances nucléaires à destination d'un pays étranger à travers un autre pays, partie à la Convention de Paris, qui a utilisé la faculté, ouverte par l'article 7 e de cette Convention, de subordonner ce transit à une augmentation du montant de la responsabilité de l'exploitant en cause.

Cette disposition propre à la Convention de Paris permet à une partie contractante, à chaque fois qu'elle le juge bon, d'exiger un montant de responsabilité — donc de garantie financière — identique à celui qui s'impose à ses exploitants nationaux. Il est à penser que bon nombre d'Etats désireront faire usage de cette faculté, car elle est seule à permettre une égalité de réparation pour tous les accidents survenant sur le territoire national, qu'il s'agisse d'un accident de réacteur, d'un accident de transport national ou d'un accident de transport en transit.

Cependant les exploitants responsables des transports risquent d'avoir quelque difficulté à répondre aux exigences tout à fait fondées du pays de transit, surtout si ce dernier a adhéré à la Convention de Bruxelles du 31 janvier 1963 complémentaire à la Convention de Paris, et a fixé le montant maximal de responsabilité à un chiffre nettement supérieur au maximum de 15 millions d'unités de compte prévu par la Convention de Paris, et qui peut atteindre 120 millions d'unités de compte. Il est évident qu'une telle garantie ne pourra être donnée sans le concours de fonds publics et il n'est pas certain que les Etats pratiquant un montant de responsabilité moins élevé seront disposés à fournir le complément de garantie

nécessaire pour que leurs exploitants soient en mesure de répondre à l'obligation qui leur est faite par le pays de transit. Sans cette garantie d'Etat, l'exploitant se verra dans l'impossibilité de faire transiter son envoi de substances nucléaires.

Une amorce de solution figure toutefois dans la Convention de Paris elle-même, au prix d'un léger amendement au texte de son article 4 d.

Cet article permet en effet à la législation d'une partie contractante de prévoir la substitution d'un transporteur à un exploitant d'installation nucléaire située sur le territoire de cette partie contractante, ce transporteur étant alors considéré comme un exploitant d'installation nucléaire située sur le territoire de ladite partie contractante, pour les accidents nucléaires survenus en cours de transport. Cette substitution nécessite l'accord du transporteur et de l'exploitant en cause.

Il suffirait que cette substitution soit autorisée également pour les exploitants d'installations nucléaires situées sur le territoire d'une autre partie contractante, avec l'accord de l'exploitant concerné qui n'aurait qu'à y gagner, pour que la responsabilité pendant le transport puisse être transférée d'un exploitant étranger ne remplissant pas les conditions de garantie exigées par le pays de transit à un transporteur ressortissant dudit pays de transit, assimilé à un exploitant nucléaire dudit pays et disposant a fortiori de la garantie en vigueur dans son propre Etat.

Exploitants nucléaires et transporteurs spécialisés auraient très certainement intérêt à examiner si cette suggestion ne pourrait pas faciliter grandement les opérations de transit dans des pays étrangers à fort montant de responsabilité, auquel cas cette question pourrait utilement être portée à l'ordre du jour de la première conférence de révision de la Convention de Paris qui doit être organisée en 1973.

TRENDS IN FEDERAL AND STATE LEGISLATION IN THE USA FOR THE PROTECTION OF THE ENVIRONMENT AND THE REGULATION OF NUCLEAR POWER PLANTS

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Abstract-Résumé-Аннотация-Resumen

TRENDS IN FEDERAL AND STATE LEGISLATION IN THE USA FOR THE PROTECTION OF THE ENVIRONMENT AND THE REGULATION OF NUCLEAR POWER PLANTS.

The ground-swell of public opinion in the USA toward preservation of the environment has resulted in a number of wide-ranging legislative proposals designed, in some cases, to effect root changes in governmental techniques of environmental protection. The electric power industry, and the nuclear power segment in particular, has felt the impact of these developments. A review of these trends is made in the context of (1) all the significant environmental considerations without regard to the source of the environmental impact and (2) the existing legislative framework, including the Atomic Energy Act, the Federal Power Act, the Rivers and Harbors Act, and the Federal Water Pollution Control Act. This discussion serves as a background for two pieces of recently enacted landmark legislation: the National Environmental Policy Act of 1969 and the Water Quality Improvement Act of 1970. The divergent approaches reflected in these two acts, and the interaction between the two, are discussed. Important organizational changes, centralizing federal pollution control responsibilities within the federal government, became effective last December under Reorganization Plan No. 3 of 1970. Among other changes, the USAEC's functions in setting general standards for radioactivity in the general environment are now performed by the new Environmental Protection Agency. During 1970 the problem of co-ordination between electric power requirements and environmental preservation received intensive study by the Office of Science and Technology. This study eventuated in a report recommending the establishment of new industry responsibilities and institutional arrangements designed to meet this problem. This is an area where additional legislative action may be anticipated. Other possible areas for new legislation, in the radiation control field, specifically, are (1) the division of responsibilities between the states and the federal government and (2) a restructuring of the review and hearing processes for facility licenses.

TENDANCES DE LA LEGISLATION FEDERALE ET DES ETATS POUR LA PROTECTION DE L'ENVIRONNEMENT ET LA REGLEMENTATION DES CENTRALES ELECTRIQUES NUCLEAIRES AUX ETATS-UNIS D'AMERIQUE

Le mouvement de fond de l'opinion publique aux Etats-Unis d'Amérique pour la préservation de l'environnement a eu pour résultat qu'un certain nombre de propositions de lois, touchant à de nombreux domaines, ont été déposées, pour modifier radicalement les techniques utilisées par le gouvernement pour la protection de l'environnement. Les industries productrices d'électricité, particulièrement dans la branche nucléaire, en ont ressenti l'impact. Une revue de ces tendances est présentée concernant: 1) toutes les considérations importantes pour l'environnement, indépendamment de la cause qui le perturbe, et 2) les fondements de la législation existante, comprenant l'Atomic Energy Act (Loi sur l'énergie atomique), le Federal Power Act (Loi fédérale sur l'énergie), le Rivers and Harbors Act (Loi sur les rivières et les ports) et le Federal Water Pollution Control Act (Loi fédérale sur le contrôle de la pollution de l'eau). Cette discussion sert de fond à deux textes de cette législation fondamentale récemment promulgués: le National Environmental Policy Act de 1969 (Loi nationale sur l'environnement) et le Water Quality Improvement Act de 1970 (Loi sur l'amélioration de la qualité de l'eau). Les points de vue divergents qui se font jour dans ces deux lois et leurs interactions sont discutés. Des changements importants d'organisation sont intervenus, qui ont pour objectif de centraliser les responsabilités fédérales du contrôle de la pollution à l'intérieur du gouvernement fédéral; ils sont entrés en vigueur en décembre dernier, en application du plan de réorganisa-

tion n°3 de 1970. Parmi ces changements, la responsabilité, jusqu' alors assumée par l' USAEC, de l' établissement des standards généraux pour la radioactivité dans l' environnement revient maintenant à la nouvelle Agence pour la protection de l' environnement. Pendant l' année 1970, le problème de concilier la demande d' énergie électrique avec la préservation de l' environnement a fait l' objet d' une étude intensive de la part du Bureau de la science et de la technologie. Cette étude est présentée dans un rapport recommandant l' établissement de nouvelles responsabilités pour l' industrie et les modifications institutionnelles proposées pour résoudre ce problème. Il s' agit d' un domaine où une action législative supplémentaire peut être prévue. D' autres champs d' action pourraient s' ouvrir à la nouvelle législation dans le domaine du contrôle des radiations, en particulier: 1) la répartition des responsabilités entre les états et le gouvernement fédéral, et 2) une restructuration de la procédure d' examen et d' audition pour la délivrance des licences d' installations.

ТЕНДЕНЦИИ В ФЕДЕРАЛЬНОМ ЗАКОНОДАТЕЛЬСТВЕ И ЗАКОНОДАТЕЛЬСТВЕ ШТАТОВ В США ПО ОБЕСПЕЧЕНИЮ ЗАЩИТЫ ОКРУЖАЮЩЕЙ СРЕДЫ И ПО УРЕГУЛИРОВАНИЮ ВОПРОСОВ, СВЯЗАННЫХ С АТОМНЫМИ ЭЛЕКТРОСТАНЦИЯМИ.

В результате усиления влияния общественного мнения в США, направленного на сохранение окружающей среды, разработан ряд обширных законодательных предположений, которые, в некоторых случаях, предусматривают коренные изменения в мероприятиях правительства по защите окружающей среды. Эти предложения оказывают свое влияние на электро-энергетическую промышленность, вообще, и на отрасль ядерной энергетики, в частности. В докладе дается обзор указанных тенденций в плане (1) всех значительных исследований окружающей среды, независимо от источника ее загрязнения, и (2) существующей системы законодательства, в которую входят Закон об атомной энергии, Федеральный закон по энергетике, Закон о реках и гаванях и Федеральный закон о контроле за загрязнением воды. Данное обсуждение служит предпосылкой для двух недавно принятых законодательных актов: Закон 1969 года о национальной политике в области защиты окружающей среды и Закон 1970 года об улучшении качества воды. Рассматриваются некоторые расхождения, сохранившиеся в этих двух законах, а также их взаимосвязь. В соответствии с Планом реорганизации № 3 от 1970 года, с декабря 1970 года приобрели законодательный характер важные организационные изменения в вопросе централизации ответственности за осуществление контроля, в рамках федерального правительства, за загрязнением окружающей среды. Среди других изменений отмечается тот факт, что в настоящее время функции КАЭ США по определению общих норм радиоактивности в окружающей среде осуществляются новым Агентством по защите окружающей среды. В течение 1970 года Управление по науке и технике интенсивно изучало проблему координации электроэнергетических потребностей и необходимости защиты окружающей среды. В результате данного изучения был подготовлен доклад с рекомендацией установить новые формы ответственности для представителей промышленности и провести организационные мероприятия, направленные на решение указанной проблемы. В этой области вполне можно ожидать принятия дополнительных законодательных актов. Другими возможными сферами применения нового законодательства в области радиационного контроля являются, в частности: (1) распределение обязанностей между штатами и федеральным правительством и (2) изменение системы рассмотрения и выдачи лицензий на установки.

TENDENCIAS EN LA LEGISLACION FEDERAL Y ESTATAL DE LOS ESTADOS UNIDOS PARA LA PROTECCION DEL MEDIO AMBIENTE Y LA REGLAMENTACION DE CENTRALES NUCLEOELECTRICAS.

La grave preocupación de la opinión pública en los Estados Unidos de América por la defensa del medio ambiente ha ocasionado a cierto número de proyectos de ley referentes a muy diversos dominios y destinadas, en algunos casos, a producir cambios radicales en las técnicas que aplican los organismos oficiales para la protección del medio ambiente. La industria eléctrica, y en particular el sector nuclear, han sentido ya los efectos de estas tendencias legislativas. La memoria lleva a cabo un examen de las mismas en relación con: 1) todas las consideraciones importantes para el ambiente, independientemente de los factores que lo perturban; y 2) el marco legislativo existente, incluyendo la Ley de Energía Atómica (Atomic Energy Act), la Ley Federal de Energía (Federal Power Act), la Ley de Ríos y Puertos (Rivers and Harbors Act), y la Ley Federal de Control de la Contaminación del Agua (Federal Water Pollution Control Act). Este estudio sirve de introducción a dos instrumentos legales de reciente promulgación: una ley general sobre la protección ambiental (National Environmental Policy Act, 1969), y otra sobre la mejora de la calidad del agua (Water Quality Improvement Act, 1970). Se estudian asimismo las divergentes tendencias reflejadas en esas dos leyes y su interacción. Se han introducido importantes cambios administrativos para centralizar las funciones de defensa contra la contaminación en el seno del Gobierno Federal, cambios que se hicieron efectivos el pasado diciembre en virtud del plan de reorganización N° 3 de 1970 (Reorganization Plan N° 3, 1970). Entre otros cambios, las funciones que competían a la USAEC en el establecimiento de normas generales relativas a la radiactividad en el medio ambiente corresponden ahora al nuevo Organismo

de Protección Ambiental (Environmental Protection Agency). Durante 1970, el problema de conciliar las necesidades de energía eléctrica y la conservación del medio ambiente fue objeto de intenso estudio por parte de la Oficina de Ciencia y Tecnología (Office of Science and Technology), estudio recogido en un informe que recomendaba el reconocimiento de las nuevas responsabilidades industriales así como las reestructuraciones institucionales necesarias para hacer frente a esos problemas. En este dominio se puede, pues, prever una acción legislativa suplementaria. Otros campos en los que cabe establecer nueva legislación con referencia al control de las radiaciones son específicamente: 1) la repartición de responsabilidades entre los Estados y el Gobierno Federal; 2) la reestructuración de los procedimientos de información, de examen y juicio contradictorio para la concesión de licencias para las instalaciones.

The ground-swell of public opinion in the United States of America towards preservation of the environment has resulted in a number of wide-ranging legislative proposals designed, in some cases, to effect root changes in governmental techniques of environmental protection. The electric power industry — and the nuclear segment in particular — has felt the impact of these developments.

The description of the year 1970 as 'the year of the environment' has already achieved wide currency in the United States. As President Nixon has stated:

"The course of events in 1970 has intensified awareness of and concern about environmental problems. The news of more wide-spread mercury pollution, late summer smog alerts over much of the East Coast, repeated episodes of ocean dumping and oil spills, and unresolved controversy about important land use questions have dramatized with disturbing regularity the reality and extent of these problems. No part of the United States has been free from them, and all levels of government — Federal, State and local — have joined in the search for solutions. Indeed, there is a growing trend in other countries to view the severity and complexity of environmental problems much as we do." [1]

The public awareness and concern have contributed to the enactment of three landmark Federal statutes during 1970 — the National Environmental Policy Act of 1969 [2], the Water Quality Improvement Act of 1970 [3], and the Clean Air Amendments of 1970 [4].

The legislative approaches discussed in this paper have, of course, been responsive to our own particular political and historical needs. They reflect the sometimes competing governmental interests found in our Federal system, as well as the present stage of our industrial development. These approaches may not be suitable for other nations, and they are offered as neither panaceas nor paradigms but merely as examples.

THE PAST LEGISLATIVE FRAMEWORK

Confronting each other in the United States today are, on the one hand, the growing public demand and need for electric energy and, on the other, the demonstrated necessity of preserving and protecting the environment. The reconciliation of those two social interests is one touchstone by which the viability of the legislative concepts discussed here may

be assessed. The impact of the 1970 legislation on the electric utility industry should first be examined in the context of the licensing framework previously in effect.

Atomic Energy Act of 1954, as amended [5]

This statute requires the grant of a construction permit from the Atomic Energy Commission (the 'AEC' or the 'Commission') before a nuclear power reactor may be constructed and the grant of an operating license before operation [6]. The Act contemplates that licensees will be subject to radiation protection standards imposed by the Commission [7]. The Commission must hold a public hearing at the construction permit stage for each nuclear power reactor whether or not there is a contest regarding the issuance of the permit. At the operating license stage, there is opportunity for a further public hearing at the request of any person whose interest may be affected by the proceeding [8]. A central purpose of these hearings is to provide an open, public review of the radiological effects of the facility on the environment.

The initial and continued control of the uses of nuclear energy by a government agency charged with the protection of the public health and safety from the radiological effects of such uses has meant that the nuclear industry in the United States has developed under a system directed more toward the prevention of accidents and uncontrolled hazards than to the correction of conditions which have led to accidents and damage to the environment. The United States system for the control of nuclear energy has had the benefit of advance planning as contrasted with primary dependence on an adversary system of law or on attempts to remedy hazardous or undesirable situations through the imposition of penalties after such situations have developed.

While the Act provides a comprehensive regime for the control of radiation effects, it does not grant to the Commission the broad 'public interest' type of charter typically associated with regulatory agencies in the United States [9]. In order to afford regulatory protection against environmental effects other than radiological effects, the Commission must look for its source of authority to a new legislation enacted in 1970.

Federal Power Act [10]

Under this Act, first enacted in 1920, the Federal Power Commission, which, like the AEC, is an independent Federal regulatory agency, licenses the construction and operation of hydroelectric plants [11]. The Act, in its inception, had as its purpose the development and preservation of the water resources of the United States [12]. In contrast to the circumscribed statutory charter for the regulation of nuclear power plants, the Federal Power Act provides the basis for comprehensive consideration and control of all environmental effects of hydroelectric plants [13].

Rivers and Harbors Act [14]

There was no comparable statutory system established for fossil-fuel plants even though those plants have provided the lion's share of the electric power generated in the United States. The Rivers and Harbors Act, first

enacted in 1899, is, however, an example of an old statute which has been reinterpreted and given new life toward the end of environmental protection. That Act requires a permit from the Corps of Engineers to build structures or conduct work in navigable waters [15]. Another provision of the Act, termed the Refuse Act, makes discharge of 'refuse matter' unlawful without a permit from the Corps [16]. Until fairly recently, implementation of the Rivers and Harbors Act had been directed primarily at the protection of navigation. Now, however, that Act is construed as serving water quality protection goals, and 'refuse matter' is construed as including thermal discharges [17]. A comprehensive regulatory program for the protection of water quality under the Rivers and Harbors Act, designed and largely developed in 1970, can be expected to make a significant contribution toward protecting the environment from the effects of the operation of electric power plants, including many fossil-fuel plants [18].

SIGNIFICANT FEDERAL LEGISLATIVE DEVELOPMENTS IN 1970

National Environmental Policy Act of 1969 ('NEPA')

This statute, enacted on January 1, 1970, is the most general, the most interesting, and perhaps the most significant, of United States statutes directed toward environmental protection. The Act makes it the continuing responsibility of the Federal Government to use all practicable means, consistent with other essential considerations of national policy, to improve and co-ordinate Federal plans, functions, programs, and resources toward certain stated ends of environmental protection [19]. It directs that, to the fullest extent possible, the policies, regulations, and public laws of the United States are to be interpreted and administered in accordance with the policies set forth in the Act [20]. All agencies of the Federal Government are required to include in every recommendation on major Federal actions significantly affecting the quality of the human environment a detailed discussion of the basic short-term and long-term environmental consequences of the proposed action [21]. Federal agencies are required to utilize a systematic, interdisciplinary approach to ensure the integrated use of the natural and social sciences in decision-making which may have an impact on man's environment [22]. They must also develop appropriate alternatives to recommended courses of action in any proposal involving unresolved conflicts concerning alternative uses of available resources [23].

The Act establishes a Council on Environmental Quality reporting to the President [24]. The Council is charged with reviewing the activities of the Federal Government in the light of the environmental policies of the Act, and of recommending national policies to foster environmental quality [25].

NEPA has had the effect of expanding the AEC's regulatory jurisdiction into the area of non-radiological environmental effects of nuclear power plant construction and operation. In its implementation of the statute, the Commission has recognized that some environmental amenities and values are presently quantified and that some are as yet unquantified. It has sought to give appropriate recognition to both categories, as well as to take into

account the traditional role played by the State and local governments in the protection of the environment [26].

Each AEC construction permit and operating license issued for a nuclear power reactor now includes a condition requiring compliance with applicable Federal and State standards and requirements for the protection of the environment [27]. This condition reflects AEC's augmented regulatory jurisdiction as well as its recognition that, under our Federal system, the preservation of environmental values can best be accomplished through the establishing of environmental quality standards and requirements by appropriate Federal, State and regional agencies having the responsibility for environmental protection. The Commission has urged the appropriate agencies to proceed promptly to establish such standards and requirements [28].

Beyond this, a new environmental issue may now be raised and litigated in AEC power reactor licensing proceedings. That issue is whether the issuance of the permit or license would be likely to result in a significant adverse effect on the environment. If such a result is indicated, consideration will be given to the need for the imposition of requirements for the preservation of environmental values consistent with other essential considerations of national policy, including the need to meet on a timely basis the growing national requirement for electric power [29].

NEPA reaches not only nuclear power plants licensed by the AEC but hydroelectric plants licensed by the FPC and all plants, both nuclear and non-nuclear, for which a permit from the Corps of Engineers is required under the Rivers and Harbors Act. The broad sweep and general language of NEPA have already engendered litigation in the Federal courts [30]. Whatever the outcome of the litigation, it seems clear that the environmental consequences of Federal decision-making must now be given appropriate anticipatory consideration.

Water Quality Improvement Act of 1970 ('WQIA')

In contrast to NEPA, this statute, enacted on April 3, 1970, is quite specific as to the administrative procedures to be used to accomplish the basic statutory objective — the control of water pollution. WQIA amended the existing Federal Water Pollution Control Act ('FWPCA') which established a framework of State-Federal co-operation under which the States were given the opportunity to set water quality standards for interstate waters. Under FWPCA, if a State fails to promulgate such standards or if the State standards are determined to be inadequate, the Federal Government may itself set the water quality standards for interstate waters within the State [31].

Experience under FWPCA pointed up the cumbersome nature of its enforcement procedures [32] as well as the desirability of providing a means of preventing pollution before it occurs rather than relying on enforcement after a violation has occurred. It was those deficiencies that give impetus and direction to the 1970 amendments.

The salient thrust of WQIA, insofar as its effect on the electric power industry is concerned, was to seize upon the leverage of Federal licensing authority so as to assure that adverse effects on the aquatic environment would be considered and dealt with beforehand [33]. Thus, the AEC, the Federal Power Commission and the Corps of Engineers, the Federal agencies

which issue permits or licenses for electric power plants, must now receive from the utility applicant, before the license may be granted, a certification that there is 'reasonable assurance' of compliance with the applicable water quality standards. The certification will ordinarily come from the State where the discharge originates though provision is made for certification, in special circumstances, by interstate agencies or by the Federal Government [34]. The requirement for certification is postponed in certain situations to afford license applicants, whose construction plans or activities are well advanced, a reasonable time to obtain the certification [35]. In an effort to accommodate the interests of other States affected by the discharge, the Congress included a rather novel provision permitting such States to object to the issuance of the license and to request a public hearing on their objection [36].

One noteworthy aspect of WQIA is its lack of complete coverage for fossil-fuel plants. This comes about because not all such plants are subject to licensing under the Rivers and Harbors Act and those that are so subject may be largely constructed before the permit from the Corps of Engineers is required [37].

Clean Air Amendments of 1970

Before enactment of these amendments on December 31, 1970, the Federal Government had authority, under the existing Clean Air Act, to establish air quality criteria for certain pollutants, but could not regulate the emission of the pollutants into the air [38]. Congress had declared air pollution control to be "the primary responsibility of state and local governments" [39]. The States were authorized to develop air quality standards and implementation plans, including emission standards, which, upon Federal approval, became the standards and plan for the air quality control region within the State [40].

The 1970 amendments gave the Federal Government a far more active role. They provided for Federal establishment of national primary ambient air quality standards (to protect health) and national secondary ambient air quality standards (more stringent standards to protect the public welfare), and an opportunity for adoption by the States of implementation and enforcement plans for such standards. Those plans must include emission limitations, provision for establishment and operation of monitoring devices, and provision for preconstruction review of the location of new stationary sources of air pollution [41]. As to new stationary sources, the Federal Government may itself establish emission standards [42]. The Federal Government may also promulgate emission standards for hazardous air pollutants from all stationary sources (whether new or existing) [43]. It must delegate to the States enforcement authority with respect to standards of performance for new sources within the State and with respect to emission standards for hazardous air pollutants. The authority of States and their political subdivisions to set standards for new stationary sources and emission standards for hazardous air pollutants is expressly reserved, provided that their requirements are no less stringent than applicable Federal standards or those contained in Federally approved implementation plans [44].

The provisions relating to new stationary sources have particular relevance for electric power plants — particularly fossil-fuel plants which release oxides of sulphur and nitrogen. As to nuclear power plants, the

legislative history of the amendments indicates that the responsibilities of the AEC with respect to radiological health and safety aspects of nuclear facilities were to remain unchanged by enactment of the amendments [45].

Reorganization Plan No. 3 of 1970

The foregoing discussion points up the proliferation of authorities responsible for various aspects of environmental protection at the Federal level. The President, in Reorganization Plan No. 3, which went into effect on December 2, 1970, consolidated some of these authorities in a new Environmental Protection Agency. Functions transferred to this Agency include, among others, administration of the Federal Water Pollution Control Act and the Clean Air Act as well as the functions of the AEC in establishing generally applicable standards for the protection of the general environment from radioactive materials [46].

The Plan reflected a recognition that our national Government was not structured to make a co-ordinated attack on the pollutants which adversely affect our environment and that, for pollution control purposes, the environment must be perceived as a single, interrelated system. Pre-existing authorities were designed primarily along media lines — air, water, and land — while the sources of air, water, and land pollution are interrelated and often interchangeable. In organizational terms, the Plan brought together in one agency a variety of research, monitoring, standard-setting, and enforcement activities to assure a more co-ordinated attack on the problems of environmental pollution [47].

OTHER LEGISLATIVE DEVELOPMENTS — FEDERAL AND STATE

The Federal statutes discussed here indicate some of the varied approaches adopted in the field of environmental protection. They are by no means a complete listing. Among other significant Federal statutes in this field are the Fish and Wildlife Coordination Act [48], which requires Federal agencies to take into account the conservation of fish and wildlife resources in connection with certain activities; the Wild and Scenic Rivers Act [49], which is designed to preserve our scenic rivers in their free flowing condition; and the National Historic Preservation Act [50], which requires Federal licensing agencies to take into account the effect of a licensed undertaking on historical sites, buildings, etc., that are included in a 'National Register'. The Solid Waste Disposal Act [51], which authorizes the expenditure of Federal funds for research and development in ways of disposing of solid wastes, was amended in 1970 to increase such funds and to emphasize recycling of solid wastes [52].

For each environmental statute enacted, there are a number of legislative proposals, with environmental protection objectives, which have not become law. Their sheer numbers and variety preclude even a summary description. They are a further reflection, however, of the public ferment and concern in this area.

Among a few disparate examples are Federal bills, introduced in 1971, which would lead to the development of a national fuels and energy policy [53]; provide for a national land-use policy [54]; provide a private right of action to protect national resources [55]; provide for a national power

plant siting study and plan [56]; regulate the discharge of materials into the oceans [57]; declare that every citizen has a fundamental and inalienable right to a healthful environment [58]; promote reliability of electric power by means which would take into account environmental considerations [59]; and permit the States to impose radiation protection standards more stringent than those of the AEC [60].

In connection with the last-mentioned proposal, it is reasonably clear that under the law as it now stands, the Federal Government has the exclusive authority to regulate, from the standpoint of protection against radiation effects, the possession and use of those facilities and materials which are subject to licensing by the AEC (in the absence of an agreement between a State and the AEC) [61]. If and to the extent that a case can be made for permitting the individual States to set their own radiation protection standards, the appropriate forum for effecting a change is the Congress where all the relevant national policy considerations can be assessed and acted upon in a coherent manner.

In addition to attempts on the part of a few States to prescribe controls on the radioactive discharges from nuclear power plants [62], there have been some sporadic, but thus far unsuccessful, State efforts to establish moratoria on the construction or operation of those plants [63]. With respect to electric power plants generally, it would seem that a far more promising target for State legislative action is the provision, within a single structure, for appropriate, timely and non-discriminatory consideration of environmental factors and energy needs. There have been significant State efforts in this direction [64].

POWER PLANT SITING - A FEDERAL PROPOSAL

One of the most far-reaching and attractive Federally-advanced concepts affecting electric power plants is found in proposed legislation to solve the problems of assuring a reliable supply of electric power to meet increasing energy demands and, at the same time, to assure that environmental protection is made an integral part of the planning, construction and operation of electric power generating facilities and high voltage transmission lines. Such a proposal, developed by the Office of Science and Technology, has been introduced in the present Congress. The bill would implement the recommendations of an inter-agency task force which studied the problem in co-operation with concerned Federal, State and local officials, industry and conservation groups, and issued a report 'Electric Power and the Environment' in October of 1970 [65].

In summary, the proposed legislation would: (1) require this Nation's electric utilities to engage in long-range planning and to publish general plans for their system expansions at least ten years in advance of construction [66]; (2) provide that each State or region may establish a decision-making body that will review alternatives in order to assure that optimum sites for power plants and large transmission lines are selected, and that adequate environmental protection features will be employed [67]; (3) provide for Federal Government review and approval responsibility until such time as a decision-making body is established on a State or regional level in accordance with Federal guidelines [68]; (4) require proposed power plant sites and general locations of transmission line routes to be disclosed

and that public hearings on the plant sites be held at least five years before construction [69]; (5) require that detailed applications be filed and another public hearing held at least two years in advance of construction [70]; and (6) provide that the decision of the State or regional power plant siting body shall be conclusive on all matters of State or local law, thus consolidating the various approvals now required at the State and local level [71].

The requirements would be applicable to all bulk power facilities regardless of ownership except that small plants and lower voltage transmission lines would be exempt [72]. After obtaining a State certificate and any necessary Federal licenses, the utilities would be authorized to begin construction using Federal powers of eminent domain if necessary to obtain the land [73].

In harmony with this legislative proposal's objective of achieving an early decision on significant site matters is a possible restructuring of the AEC's licensing and hearing process that has been adumbrated by AEC Commissioner James T. Ramey [74]. This restructuring, if carried out, would provide for an early determination of nuclear power plant site suitability after consideration of the appropriate environmental factors in a public forum. This determination would be made before any clearing of land or other action that would adversely affect the natural environment of a site.

CONCLUSION

The massive and varied legislative response to present-day environmental concerns in the United States is not yet over. That response has brought about a proliferation of authorities responsible for various aspects of environmental protection. Part of the duplication and overlap can be attributed in part to our Federal system of shared sovereignty, but there has been a discernible trend toward consolidation of authorities at the Federal level as well as a developing pattern of Federal-State co-operation.

It is now clear that major Federal decisions must take into account the environmental consequences of the proposed action and that those consequences must be balanced against other essential considerations of national policy. The trend is toward more advance and open planning in the location and construction of potential sources of environmental pollution. This trend reflects the lessons of the situation in which we now find ourselves with respect to the results of many of our technological and industrial advances. Advance planning not only helps to avoid the costs involved in restoring an environment polluted as a result of unplanned activities, but also permits wider public participation in the planning process.

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- [3] 84 Stat. 91.
- [4] 84 Stat. 1676.
- [5] 68 Stat. 919, 42 U.S.C. 2011 et seq.

- [6] 42 U.S.C. 2131, 2235.
- [7] 42 U.S.C. 2073(e)(7), 2093 (b), 2111, 2133 (b), 2134, 2201(b), (i).
- [8] 42 U.S.C. 2239(a).
- [9] State of New Hampshire v. Atomic Energy Commission, 406 F. 2d 170 (1st Cir., 1969), cert. den., 395 U.S. 962 (1969).
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- [14] 33 U.S.C. 401-415.
- [15] 33 U.S.C. 403.
- [16] 33 U.S.C. 407.
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- [19] 42 U.S.C. 4331(b).
- [20] 42 U.S.C. 4332(1).
- [21] 42 U.S.C. 4332(2)(C).
- [22] 42 U.S.C. 4332(2)(A).
- [23] 42 U.S.C. 4332(2)(D).
- [24] 42 U.S.C. 4342.
- [25] 42 U.S.C. 4344.
- [26] Title 10, Code of Federal Regulations, Part 50, Appendix D.
- [27] Id., paragraph 11(c).
- [28] Id., paragraph 10.
- [29] Id., paragraph 11(a).
- [30] E.g., Wilderness Society v. Hickel, 1 E.R. 1335 (D. D. C., 1970) Texas Committee v. U.S., 1 E.R. 1303 (W.D. Tex., 1970); Calvert Cliffs' Coordinating Committee, Inc. v. U. S. Atomic Energy Commission, Docket #24,839 (D.C. Cir.).
- [31] 33 U.S.C. 1160(c)(1)-(4).
- [32] 33 U.S.C. 1160(d)-(g).
- [33] WQIA contains many other provisions to protect water quality relating to such matters as oil pollution, hazardous materials, and marine sewage.
- [34] 33 U.S.C. 1171(b)(1).
- [35] 33 U.S.C. 1171(b)(7) and (8).
- [36] 33 U.S.C. 1171(b)(2).
- [37] Of twelve large fossil-fuel plants built in 1967, only seven required a Corps of Engineers permit. Application for a Corps of Engineers permit need not be made until construction of the intake and discharge structures.
- [38] 81 Stat. 490.
- [39] 77 Stat. 392.
- [40] 81 Stat. 491.
- [41] 84 Stat. 1679-1682.
- [42] 84 Stat. 1683-4.
- [43] 84 Stat. 1685.
- [44] 84 Stat. 1689. The Clean Air Act Amendments contain other provisions dealing with motor vehicle emission standards, regulation of fuels and fuel additives, and aircraft emission standards.
- [45] H.R. Rep. No. 91-1783, 91st Cong., 2d Sess. (1970); 116 Cong. Rec. S20610 (daily ed., Dec. 18, 1970); 116 Cong. Rec. H12060 (daily ed., Dec. 18, 1970).
- [46] The Environmental Protection Agency has published regulations under the transferred authority -- e.g., proposed regulations under the Federal Water Pollution Control Act, 36 Fed. Reg. 2516, Feb. 5, 1971; proposed regulations under the Clean Air Act, 36 Fed. Reg. 1502, Jan. 30, 1971.

- [47] See President Nixon's Message to Congress on Reorganization Plan No. 3 on July 9, 1970, 116 Cong. Rec. S10876.
- [48] 16 U.S.C. 661 et seq.
- [49] 16 U.S.C. 1271-1287.
- [50] 16 U.S.C. 470-470m.
- [51] 42 U.S.C. 3251 et seq.
- [52] 84 Stat. 1127.
- [53] E.g., S. Res. 45. This resolution and the bills referred to in references 54-60 were introduced in the 92nd Cong., 1st Sess.
- [54] E.g., S. 632.
- [55] E.g., H.R. 2288.
- [56] E.g., H.R. 1079.
- [57] E.g., H.R. 5239.
- [58] E.g., S. 3575.
- [59] E.g., H.R. 3838.
- [60] E.g., H.R. 3683.
- [61] 42 U.S.C. 2021; Northern States Power Company v. State of Minnesota, 320 F. Supp. 172 (1970).
- [62] E.g., Minnesota, where the State's attempt to prescribe such controls has resulted in litigation in the Federal Courts (see ref. 61).
- [63] E.g., Oregon (Senate Bills 51, 218).
- [64] E.g., New York - Ch. 272, Sess. L. of 1970, Sec. 6.
- [65] The interagency task force had previously issued a report "Considerations Affecting Steam Power Plant Site Selection", dated December 1968. The Congressional Joint Committee on Atomic Energy held hearings on "Environmental Effects of Producing Electric Power" at the end of 1969 and the beginning of 1970.
- [66] Sec. 4.
- [67] Sec. 5(a), (b).
- [68] Sec. 5(c)-(e).
- [69] Sec. 4(a)(2).
- [70] Sec. 6
- [71] Sec. 7(a).
- [72] Sec. 3(a)-(c). Hydroelectric plants licensed under the Federal Power Act would not be subject to the certification requirements and the provisions relating to approval of proposed sites.
- [73] Sec. 10.
- [74] Commissioner Ramey has discussed this possible restructuring on several occasions, including remarks on "Reactor Licensing Delays in Perspective: Problems, Progress, New Directions" before the Atomic Industrial Forum Annual Conference, Washington, D.C., November 17, 1970.

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THE ROLE OF CODES AND STANDARDS IN ACHIEVING SAFE, DEPENDABLE AND ECONOMIC NUCLEAR POWER

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Abstract-Résumé-Аннотация-Resumen

THE ROLE OF CODES AND STANDARDS IN ACHIEVING SAFE, DEPENDABLE AND ECONOMIC NUCLEAR POWER.

The rapid expansion of nuclear power in the USA has placed heavy burdens on every sector of the industry, supplier, architect-engineer, constructor, and utility, as well as on the regulatory authority, the USAEC. Early in this period of rapid growth it became evident that this burden could be markedly alleviated if an ample body of nuclear codes and standards was available. Action was taken to expedite standards development on two broad fronts: (1) within the nuclear industry, by greatly increasing the tempo of nuclear standards work under the aegis of the American National Standards Institute (ANSI); and (2) within the USAEC, by encouraging and collaborating with this work and by establishing programs to develop specification-type standards for procurement of components and services for government-owned research and development facilities and to develop safety criteria and guides for commercial facilities. The traditional American system of standards development is voluntary, with the work being done by knowledgeable people who contribute their time. Briefly, the ANSI designates technical societies and industry associations as secretariats and sponsors for specific standards-writing assignments. The paper describes the current nuclear standards program, in which over 1000 people are engaged, with particular emphasis on organization and procedural features designed to expedite standards production. It includes the standards organizations of the technical societies that do the vast bulk of standards drafting and updating work. Statistics showing the number of workers supplied by various sectors of industry and government are included. Methods by which priorities are established and progress in the principal priority efforts are indicated. The various ways in which the USAEC is supporting the program are also described. The contribution of good standards to sound engineering and quality construction is featured in the context of economics. A lost day of operation for a nuclear power station may cost as much as \$100 000 a day. Thus the expense of developing standards that reduce this potential loss can be easily justified.

LE RÔLE DES CODES ET DES STANDARDS POUR L'OBTENTION D'UNE ÉNERGIE NUCLEAIRE SURE, FIDÈLE ET ÉCONOMIQUE.

Le développement rapide de l'énergie nucléaire aux États-Unis d'Amérique a imposé une lourde charge à chaque secteur de l'industrie: fournisseurs, compagnies d'ingénierie, constructeurs, services publics, ainsi que sur l'autorité coordinatrice, l'USAEC. Au cours de cette période de croissance rapide, il est bientôt devenu évident que cette charge pourrait être sensiblement allégée si un vaste ensemble de codes et de standards nucléaires était disponible. Une action a été entreprise sur deux fronts principaux pour hâter la définition des standards: 1) l'industrie nucléaire a beaucoup augmenté le rythme de travail sur les standards

nucléaires sous l'égide de l'ANSI (Institut national américain des standards), et 2) l'USAEC a encouragé ce travail et y a collaboré en établissant des programmes pour définir des standards types, en vue de la fourniture de composants et de services destinés aux recherches dirigées par le gouvernement et aux installations de développement, et pour définir des critères et des guides de sécurité dans les installations commerciales. Le système américain traditionnel pour la mise au point des standards est un effort spontané, résultant de la collaboration de beaucoup de personnes qualifiées qui y consacrent une partie de leur temps. En bref, l'ANSI désigne des sociétés techniques et des associations industrielles pour assurer le secrétariat et la direction des tâches de définition de standards spécifiques. Le mémoire décrit le programme actuel de standards nucléaires, auquel travaillent plus de 1000 personnes. L'accent est plus particulièrement mis sur les caractéristiques de l'organisation et des procédés mis en place pour accélérer la production de standards. Les divisions des sociétés techniques, entre autres, ont la lourde tâche de définir et de moderniser les standards. Des statistiques montrant le nombre de travailleurs fournis par les divers secteurs de l'industrie et le gouvernement sont incluses. Les méthodes permettant d'établir les priorités ainsi que les progrès réalisés dans les principaux domaines prioritaires sont mentionnés. Les différentes voies que l'USAEC utilise pour soutenir le programme sont aussi décrites. L'existence de standards de qualité est une des conditions indispensables pour assurer un déroulement satisfaisant des phases de projet et de construction et les caractéristiques de leur contribution sont indiquées dans le contexte économique. Un jour de fonctionnement perdu pour une centrale nucléaire peut coûter jusqu'à \$100 000. Ainsi peuvent être facilement justifiées les dépenses engagées pour l'élaboration de standards qui réduisent les possibilités de telles pertes.

РОЛЬ КОДОВ И СТАНДАРТОВ ДЛЯ ОБЕСПЕЧЕНИЯ БЕЗОПАСНОСТИ, НАДЕЖНОСТИ И ЭКОНОМИЧНОСТИ РАЗВИТИЯ ЯДЕРНОЙ ЭНЕРГЕТИКИ.

Ускоренное развитие ядерной энергетики в США привело к дополнительным нагрузкам в различных отраслях промышленности, в системе снабжения, в строительном деле, в проектных работах, а также сделало работу руководящего органа — Комиссии по атомной энергии США — более интенсивной. В начальный период ускоренного развития ядерной энергетики стало уже очевидно, что эти нагрузки могут быть значительно сокращены, если будет разработана широкая система ядерных кодов и стандартов. Были предприняты меры по скорейшей разработке стандартов по двум широким направлениям: (1) в атомной промышленности — путем интенсификации работ по выработке ядерных стандартов под руководством Американского национального института стандартов (АНИС); и (2) в КАЭ США — путем финансирования и оказания помощи при проведении указанных работ, а также путем включения программ по разработке стандартов в виде спецификаций для закупки отдельных видов оборудования и оказания услуг государственным научно-исследовательским и опытно-конструкторским учреждениям и, наконец, путем разработки критериев безопасности и основополагающих директив для коммерческих учреждений. Традиционная американская система разработки стандартов носит добровольный характер, причем эта работа выполняется компетентными специалистами, которые жертвуют своим личным временем. Коротко говоря, деятельность АНИС направлена на создание технических обществ и промышленных ассоциаций, которые служат в качестве секретариатов и заказчиков на составление различных стандартов. В докладе описана программа по современным ядерным стандартам, в разработке которой занято свыше 1000 человек; основное внимание в докладе уделено организационным и процедурным мероприятиям, предназначенным для ускорения работ по выработке стандартов. Описаны отделы стандартизации технических обществ, которые выполняют большой объем работ по подготовке проектов стандартов и по приведению стандартов в соответствие с современными данными. Приведены статистические данные по количественному составу сотрудников, выделенных для указанных работ различными секторами промышленности и государственными учреждениями. Обсуждаются методы установления приоритета и показываются достижения в деле установления приоритета в области стандартов. Описаны также различные применяемые КАЭ методы содействия в осуществлении данной программы. С точки зрения экономики рассматривается влияние обоснованных стандартов на создание технически надежных и высококачественных сооружений. Суточный простой атомной электростанции может привести к убыткам порядка 100 тыс. долл. в день. Поэтому затраты на разработку стандартов, которые могут привести к сокращению таких возможных убытков, вполне оправданы.

IMPORTANCIA DE LOS REGLAMENTOS Y NORMAS PARA OBTENER ELECTRICIDAD NUCLEAR EN CONDICIONES DE SEGURIDAD, REGULARIDAD Y ECONOMIA.

La expansión rápida de la energía nucleoelectrica en los Estados Unidos de América ha supuesto una pesada carga para todos los sectores de la industria: proveedores, ingenieros y arquitectos, constructores y compañías eléctricas, así como para la autoridad reglamentadora, la USAEC. Ya en los comienzos de este período de rápido crecimiento se advirtió que dicha carga podía aligerarse notablemente si se conseguía crear una amplia colección de reglamentos y normas en materia nuclear. Se han tomado medidas para acelerar la

redacción de normas en dos amplios sectores: 1) en el marco de la industria nuclear, aumentando considerablemente el ritmo de elaboración de normas bajo el patrocinio del American National Standards Institute (ANSI); 2) en el marco de la USAEC, estimulando y contribuyendo a esta tarea, así como emprendiendo programas de elaboración de normas (tipo pliegos de condiciones) para la obtención de equipo y servicios con destino a las instalaciones nacionales de investigación y desarrollo, además de programas encaminados a definir los criterios y directrices aplicables en materia de seguridad a las instalaciones industriales. Con arreglo al sistema tradicional americano, la elaboración de normas tiene carácter voluntario, corriendo a cargo de expertos que ofrecen desinteresadamente su tiempo. En síntesis, el ANSI designa a las sociedades técnicas y asociaciones industriales que realizan los trabajos de secretaría y promoción necesarios para redactar las normas de que se trate. La memoria describe el programa actual de normas nucleares, para el que trabajan más de 1000 personas, prestando especial atención a las medidas de organización y procedimiento encaminadas a acelerar la elaboración de normas. Muestra cómo están organizados los servicios de las sociedades técnicas que realizan la gran mayoría de los trabajos de redacción y puesta al día de normas. Se incluyen estadísticas que indican el número de personas dedicadas a esta actividad en varios sectores de la industria y de la Administración. Se exponen los métodos aplicados para la atribución de prioridades, así como el desarrollo de los principales trabajos prioritarios. También se indican las diversas formas en que la USAEC contribuye al programa. En el aspecto económico, la elaboración de buenas normas, que garanticen trabajos de ingeniería y construcción de elevada calidad, tiene una gran trascendencia. Un solo día de interrupción del funcionamiento de una central nuclear puede suponer una pérdida de 100 000 dólares. Quedan, pues, perfectamente justificados los gastos de preparación de normas que reduzcan la posibilidad de tales pérdidas.

INTRODUCTION

The rapid expansion of nuclear power in the United States of America has placed heavy burdens on every sector of the industry – supplier, architect-engineer, constructor, and utility – as well as on the USAEC regulatory activity and development programs. Early in this period of rapid growth it became evident that the availability of an ample body of adequate nuclear codes and standards would result in significant improvements in safety and reliability, in stating and meeting requirements and in facilitating licensing reviews. Further, failure to take advantage of these benefits can result in construction delays and a consequent financial penalty in excess of US \$50 000 000 per year on an 1100 MW(e) plant.

Action was taken to expedite standards development on two fronts: (a) within the nuclear industry by greatly increasing the tempo of nuclear standards work under the American National Standards Institute (ANSI); and (b) within the USAEC by encouraging and collaborating with this work and by establishing programs to develop standards for procurement of components and services for government research and development facilities and to develop safety criteria and guides for commercial facilities.

The role of codes and standards in achieving safe, dependable, and economic nuclear power is discussed within the framework of the following areas: (a) Industrial standards development; (b) Role of the USAEC; (c) Importance of codes and standards to the utility owner.

INDUSTRIAL STANDARDS DEVELOPMENT

Traditionally, the United States have developed their technical standards through the voluntary method. Many competent men donate their time in writing and reviewing these standards. When the typical standard has been through the prescribed series of evaluations and revisions by whatever task

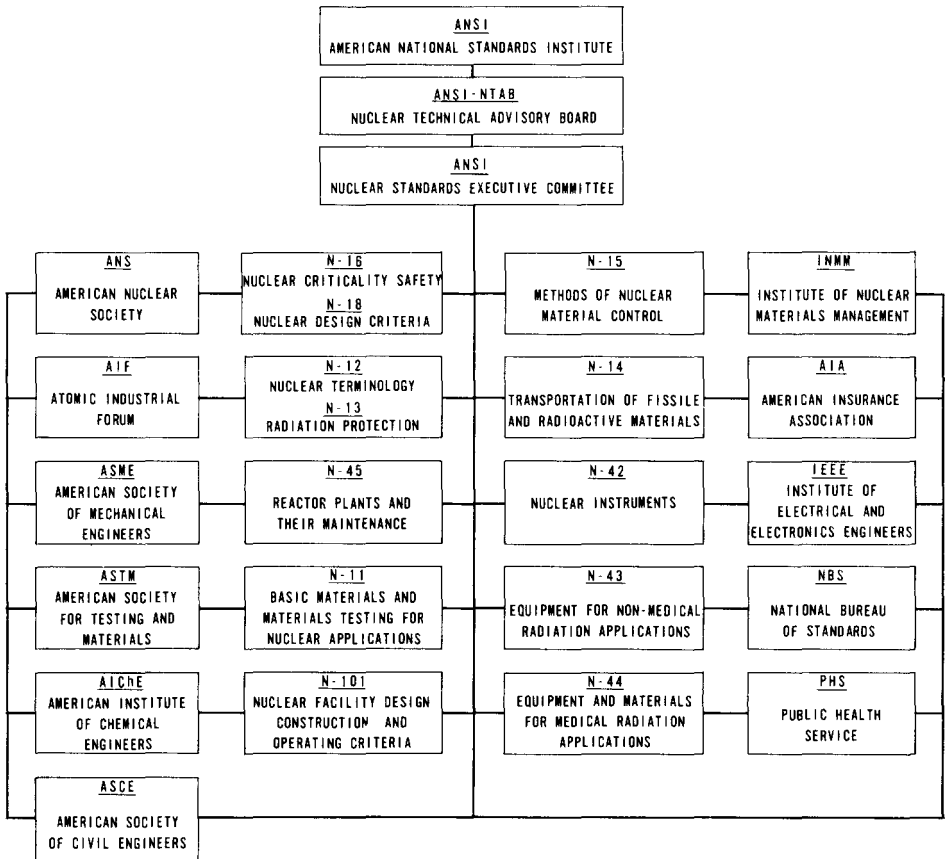


FIG. 1. ANSI Nuclear Standards Organization.

groups, sub-committees, or committees the sponsoring organization feels are necessary, it may then be adopted by that organization. If it is to be accepted nationally, it goes through another, broader series of examinations under the auspices of the American National Standards Institute (ANSI) to ensure that all interested parties have had an opportunity to comment on its adequacy and pertinence.

This system precludes setting standards that may favour the product of one manufacturer over that of another. No one organization or group has a preferential position because all interested parties are consulted and have the opportunity to comment or appeal.

Although many nuclear standards have been produced by this system, the tremendous growth that the nuclear industry is experiencing in the United States has created an urgent need for a great many more. Millions of dollars could be saved by utilities, manufacturers, and the government,

in the nuclear industry as it could in other large-scale energy applications, by eliminating duplication, errors, and uncertainty occasioned by lack of approved standards of design practice. If there were guiding standards, these organizations would know what is required in any given area and would not have to expend the great amount of effort that now goes into parametric studies, parallel solutions, methods development, and providing justification for each aspect of the design.

Figure 1 shows the ANSI Nuclear Standards Organization including the designation and technical areas of the various ANSI Standards Committees (N-Committees) and their sponsoring societies.

A brief outline of the method of preparing standards in the American Nuclear Society [1] will illustrate a method that is typical of that used by many similar organizations in the United States.

There are four levels in the ANS (American Nuclear Society) development of a standard: Individual, Working Group, Sub-committee, and Segment of the Society. The Individual is the person selected to write the proposed standard. The Working Group is the primary review board. The writer and the members of the Working Group (from four to eight) are selected for their professional competence in the area, or areas, involved. The Sub-committee consists of several Working Groups operating in related fields. The Segment of the Society is composed of selected specialists, the Standards Committee, the Board of Directors, and the officers of the Professional Division whose scope encompasses that of the proposed standard. Thus, the development process advances from an individual to a small group of approximately five, to a sub-committee of ten, and finally to a major portion of the Society consisting of possibly a hundred people.

Tracing the course of a new standard, one can outline the following steps:

(a) A writer (Individual) prepares the standard. It is then passed on to the Working Group which reviews, comments on and returns the standard to the writer for revision. This review-revise procedure is repeated until all members of the Working Group and the writer are satisfied and have approved the standard.

(b) The partially approved standard is submitted to the Sub-committee. The review and revision process now involves the Sub-committee, the Working Group and the Individual. This procedure is repeated until approval is finally obtained at all three levels.

(c) The Segment of the Society is the final body to receive the new standard and again the review-revise iteration occurs. Changes recommended at this highest level are studied at the third, second, and first levels, and when approval is obtained at all four levels, the standard is submitted to the Standards Committee for formal adoption.

(d) The Standards Committee exercises its judgement by vote of each person on the Committee. Final disposition is based on the consensus principle; that is, the votes are weighed rather than counted. If negative arguments are deemed valid, the standard is returned to the Sub-committee for additional work. If there are no negative arguments, or if those presented are deemed invalid or inconsequential, the standard is approved and submitted to the Board of Directors. The Board checks the procedures that were followed in producing the standard and sends it to the President for signature if the procedures were as specified by the Society.

It is obvious that this procedure provides each participant at any of the four levels with ample opportunity to contribute in the field of his professional competence. The result is a high probability that a new standard will be adopted without further revision.

If an organization originating a new nuclear standard wants it to become an American National Standard, it is submitted to the American National Standards Institute. While the technical content of a standard is best judged by the organization that undertakes its development, nationwide applicability of a standard is best judged by ANSI. Standards submitted to ANSI are subjected to very broad exposure and critical examination during a period of public review and comment. Through the procedures of the independent 15-member Board of Standards Review of ANSI, any interested or affected party, public or private, has an opportunity to voice an objection to a proposed standard before its final acceptance. In this way oversights, inadequacies and mistakes are corrected and conflicts are resolved [2]. The result, invariably, is a much stronger and more germane standard.

In a move to expedite the production and acceptance of nuclear standards, ANSI now permits the submittal of candidate standards directly from the Sub-committee level of national organizations such as ANS.

No matter what the history of a candidate standard, it goes first to one of the twelve ANSI Standards Committees (N-Committees) set up by the Nuclear Technical Advisory Board (NTAB). The members of each N-Committee represent a cross-section of all organizations having a substantial interest in the nuclear areas covered. Each member reacts to the proposed standard by letter ballot in which he expresses his opinion as to the preparation of the standard and its adequacy. The Secretariat of the N-Committee goes over the votes and decides if a consensus exists. If it does not, the standard goes back to the originating organization and the N-Committee works with that organization to resolve the difficulty. When the standard is approved, it goes on to the Board of Standards Review.

The Board of Standards Review publishes in 'Standards Action', a bimonthly ANSI publication, the title and number of the standard, the time limit for comment, information required to obtain the standard for study, and a short description. The public is given no more than 120 (usually 60) days to submit comments. Concurrently the standard and information on the voting record of the committee involved in the preparation of the standard are sent to each member of the NTAB to be reviewed. Comments from the public and the NTAB are returned to the Board of Standards Review and then forwarded to the N-Committee for evaluation. If revisions are necessary, the standard is again returned to the originating organization and the iterative cycle repeated until final approval is obtained from the Board of Standards Review.

The function of the Board of Standards Review is the judicial one of verifying that the following requirements have been met:

- (a) Views of all substantially concerned parties have been carefully considered.
- (b) Potential use of the proposed standard has been determined.
- (c) Significant conflict with other national standards has been resolved.
- (d) Consideration has been given to other standards having national acceptance in the field.

- (e) Accord with the public interest has been assured.
- (f) Unfair provisions have been deleted.
- (g) Technical quality has been determined.
- (h) ANSI procedures have been followed.

If 10 of the 15 members of the Board of Standards Review approve, the proposed document becomes an American National Standard.

Although the voluntary standards procedure is often tedious and protracted, it is the most equitable and effective method available today. Because standards development is carried on largely by experts or at least by knowledgeable members of the technical communities most directly concerned, the probability of major revision and iterative recycle is usually not large. Also, no other method provides as much protection for the ultimate consumer.

The United States nuclear standards program is administered by the 66-member NTAB. The Board takes responsibility for general administration and co-ordination of standards preparation and approval, identifies areas in which standards should be developed, sets priorities, organizes and disbands N-Committees, surveys all subjects under consideration to avoid duplications of effort, and reviews proposed standards for technical competence.

A priority listing of the 78 most needed nuclear standards has been compiled by the Special Task Group of the NTAB. The ten most urgently needed standards are (no priority indicated by the order used):

- (1) Seismic Siting and Design Criteria
- (2) Pressurized Water Reactor Supplementary Criteria
- (3) Boiling Water Reactor Supplementary Criteria
- (4) Training and Qualifications for Nuclear Power Plant Operating Staff
- (5) Reactor Coolant Pressure Boundary Leak Detection System Design
- (6) Protection System Criteria Covering Common Mode (Systematic) Failures, Actuators and Sensing Lines
- (7) Preoperation and Startup Testing Requirements for Nuclear Power Plants
- (8) Periodic Testing Requirements for Nuclear Power Plants
- (9) Containment Isolation System Design
- (10) Quality Assurance Criteria and Practices for Nuclear Power Plants

The 'PWR Supplementary Criteria' and the 'Training and Qualifications for Nuclear Power Plant Operating Staff' standards are nearing final approval. As these and other standards in the 'top ten' are approved, they will be replaced by the highest priority items in the remaining 68. Other standards will be added to the overall list as they are identified. All of the current 78 priority standards are being examined by the Special Task Group. This is being done to prevent overlapping of efforts and to alter the scope of the standard if necessary.

There are now some 1200 technical people, representing about 400 organizations, working on nuclear standards in the United States. According to a recent survey by the Special Task Group [3], 102 come from utility companies, 298 from the government, 535 from industry, and more than 250 from other organizations. This may seem like a sufficient supply of

talent, but it is not. It is estimated that over 1500 nuclear standards must be produced in the next decade. To meet this challenge, about 50 new standards-writing operations should be initiated within the next year.

The development and maintenance of standards is a continuing process. Usually, the committees that develop codes and standards continue to function after the standard is published. They are available for interpretation of specific applications of the standard which automatically brings about its continual review, and for settling differences between industry and the regulatory agencies or between an owner and a supplier. Thus, their expertise remains effective indefinitely, and assures that the current standard reflects up-to-date experience and technology.

ROLE OF THE UNITED STATES ATOMIC ENERGY COMMISSION

Basic regulations

The enactment of the Atomic Energy Act in 1954 in the United States signalled the beginning of the development of the United States civilian nuclear program. Early in this program the United States Atomic Energy Commission (USAEC) recognized that technically sound criteria, codes, and standards for the design, construction, and operation of nuclear power plants would be needed and developed general regulations for the licensing of nuclear reactors. These regulations are contained in Title 10 of the Code of Federal Regulations (10 CFR) Part 100, the principal AEC regulation covering evaluation of the suitability of power reactor sites, Part 20, the limits of personnel exposure and limits for the release of radioactive materials under normal conditions, and Part 50, the general requirements for the design, construction, and operation of nuclear power reactors.

General design and quality assurance criteria

In 1965, a significant addition to the Commission's regulations was proposed with the development of the General Design Criteria (GDC) for light-water-cooled nuclear power plants. Comments and suggestions received on the original 27 criteria and further technical developments led to the publication in 1967 of a revised set of criteria. Based on comments received on these revised criteria and subsequent developments in technology and in the licensing process, the USAEC on February 20, 1971, adopted General Design Criteria for Nuclear Power Plants as Appendix A to 10 CFR, Part 50, of the Commission's regulations. The purpose of the adopted GDC is to establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the AEC. The GDC also provide guidance in establishing the principal design criteria for other types of nuclear power plants.

Another important addition to the USAEC regulations resulted from the development of the 'Quality Assurance Criteria for Nuclear Power Plants'. These criteria were developed in response to overwhelming evidence in early plant construction and operation that one of the more pressing needs in the nuclear power industry, both from a safety and operational point of view, was for more stringent quality assurance

practices. The Quality Assurance Criteria were adopted by the AEC in July of 1970 as Appendix B of 10 CFR, Part 50, of the Commission's regulations. Special efforts are now being made by the AEC through its compliance function to assure that these criteria are being implemented in the design, construction, and operation of current plants.

Other USAEC criteria

The USAEC develops other important criteria for the design, construction, and operation of nuclear power plants. For example, the USAEC recently adopted an amendment to 10 CFT, Part 50, which sets forth minimum requirements for plans to cope with radiation emergencies at power reactors. The amendment also provides guidance for the development of adequate emergency plans.

Criteria now under development by the USAEC that should be published for public comment in the near future include primary reactor containment leak rate testing criteria, fracture toughness criteria for reactor coolant system components, seismic and geologic siting criteria for nuclear power plants.

Safety guides

To aid applicants and licensees in complying with the general guidance provided in the Code of Federal Regulations, including the guidance found in the 'General Design Criteria for Nuclear Power Plants', the USAEC develops documents which are referred to as Safety Guides. Thirteen Safety Guides have been published to date. The primary purpose of these safety guides is to make available acceptable methods of implementing the USAEC regulations and solutions to other safety issues related to water-cooled nuclear power plants that are acceptable to the USAEC. Safety guides are not substitutes for regulations. The guides are used to describe acceptable solutions to safety issues in cases where it has not been determined that a particular solution should be made a Commission requirement.

Industry standards

In addition to the efforts of the USAEC to promote the development of standards, the nuclear industry in the United States has an active standards development program which has been previously described. A significant number of the USAEC staff are members of the industry code committees that are actively engaged in nuclear standards development. Through this participation, the technical societies and engineering code committees are kept informed of USAEC requirements, criteria and standards. Conversely, staff members are exposed to the technical expertise of a broad segment of the United States nuclear industry. Thus, both the USAEC and the nuclear industry can work together to develop an ample body of nuclear codes and standards.

An illustration of the importance that the USAEC places on adequate industry codes is a proposed amendment to Part 50 of the USAEC regulations called 'The Codes and Standards Rule'. This rule would require that a number of recently developed industry nuclear codes be followed as minimum regulatory requirements. These industry codes include the ASME Codes

for Nuclear Pressure Vessels, Nuclear Piping, Nuclear Pumps and Valves, and In-service Inspection. The proposed rule would also require that the Criteria for Reactor Protection Systems of the Institute of Electronics and Electrical Engineers be followed for all water-cooled nuclear power plants. Most comments received on the proposed change in the regulations have been favourable. However, a modified approach has been adopted for determining the date by which the more recent editions of these industry codes must be followed. It is expected that the revised Codes and Standards Rule will be adopted by the AEC later this year.

Division of reactor development and technology standards program

The need and value of standards for commercial nuclear power applications is easily understood, but their effectiveness as a tool for research and development purposes is frequently underrated. All new technology builds on the old technology of established systems. Unless this is done, much of the effort will be dissipated in trying to relearn what has already been thoroughly established. In the liquid-metal fast breeder program, the use of standards to assure effective use of established technology is of particular importance because progress in the unexplored areas of the breeder technology is already hampered by funding restrictions. Hence, one has to use as much of the existing technological base as is possible in order to attain the end purpose quickly and efficiently.

For example, the new fast breeder reactor technology will make use of established standards, and it will build on this technological base in order to develop a successful breeder. Consider, for example, the use of stainless steel alloys as the fabrication material for the LMFBR primary coolant system. These materials are almost identical to some of those used in many of the water-cooled systems applications, where there are established techniques for fabrication and quality control. Through the use of the recognized standards in these applications, one is able to define the level of quality that will result by reference to existing water-cooled reactor systems. This then is the point of departure from which the new technology must evolve. In the LMFBR program, these materials will be used at elevated temperatures where creep is a major structural consideration. Therefore, one can work from a known base to determine what modifications will be required in available material to improve creep characteristics while still using the recognized standards of practice. Alternatively, one can use this base as a means of defining the metal creep characteristics of existing materials if one elects to use these as the controlling engineering criteria in the breeder application.

A second important use of standards is to correlate successful experiments with understood practices. The development of a satisfactory fuel element for the LMFBR is one of the most urgent needs. Once one has attained success, the next matter of urgency will be to assess the economic impact of the fuel costs for the breeder power economy. One will then want to know wherein the practices for the successfully developed fuel correspond with known manufacturing costs in established nuclear power systems. Therefore one must have a correlation with the established practices of the on-going nuclear power technology. The economic appraisal will be validated by referencing the standards of engineering practice used in the manufacture of metal-clad oxide fuels which are well

established for water-cooled nuclear power systems. It is not necessarily anticipated that the practices for one system will be exactly the same as those for another, but one must know wherein they differ and how those differences will affect the economics. One must know the practicalities of the existing fuel manufacturing techniques if one is to avoid the pitfalls of unrealistic requirements for the successful breeder fuels. It is the use of this reference technology which standards define that enables one to make economic sense out of any new technological developments.

In the USAEC Reactor Development and Technology (RDT) program emphasis has been placed on the definition of standards to be used in the research and development work. Attention is being devoted to both the basic standards existing in the water-cooled reactor systems and the development of new standards for liquid-metal systems. The Oak Ridge National Laboratory operated by the Union Carbide Nuclear Division and Liquid Metal Engineering Center operated by the Atomics International Division of North American-Rockwell have, through the RDT Standards Program, assumed the task of integrating the standards information. In addition, every active experimental installation of the USAEC Reactor Development Program is participating in the definition, preparation, and application of such standards in order to make certain that the technological base is understood and used to the maximum advantage. To date over 100 standards have been issued and over 200 are being prepared.

Although the thrust of the standards effort for research and development purposes has been mainly to attain success for the reactor development and technology program, there has been an important interaction between this effort and the separate interest of the USAEC in standards aimed at public safety. During the study and development of the standards pertinent to the research and development effort, there has been an effective interchange of information with the US Nuclear Industry and the regulatory staff of the USAEC that has been valuable in reaching a common understanding concerning the established practices used in nuclear power systems. The effort directed toward the research and development interests has stimulated the parallel effort within the US Nuclear Industry to provide sharper definition of the standards on which the industrial success is founded.

IMPORTANCE OF CODES AND STANDARDS TO THE UTILITY OWNER

In the United States, the owner of a nuclear power station is responsible for its design, procurement of manufactured items, construction, quality assurance, testing, start-up, operation and maintenance. The owner also is the applicant before regulatory agencies for necessary licenses and permits. In these multiple roles, the availability of high quality codes and standards covering important features of a nuclear power station is of great importance to owners for several reasons.

A standard provides a basis of uniform communications to the many organizations and individuals performing these several functions leading to a successful nuclear power station. In the absence of standards defining minimum necessary nuclear quality, the uncontrolled results of independent and non-uniform engineering decisions as to the quality to be achieved would result in an array of qualities, some below accepted minima and

some unnecessarily expensive. It should be pointed out, however, that even the existence of an excellent standard can be no substitute for the continued application of sound engineering judgement.

Having standards available from the inception of a project is important to the owner with respect to costs and schedules. In the earlier days of nuclear power, there were instances where the absence of a standard in the early phases of a project caused major modifications and back-fitting late in the schedule. Had a standard been available, the work could have been done right the first time with obvious cost savings. The consequent delays to schedule can also have enormous cost perturbations. Until a plant goes into service, it cannot produce revenue, and interest on the construction funds for a 1100 MW(e) nuclear unit is about US \$40 000 per day. Further, when a nuclear unit is operable, it displaces energy that otherwise would be generated by older, less efficient fossil plants. The incremental difference in fuel cost can be as much as 4 mills/kWh. Thus, each day's delay in completion of a 1100 MW(e) nuclear unit could cause additional fuel expense up to US \$100 000. Therefore, the cost of developing standards that can and do reduce the exposure to delays can be easily justified.

Since most states within the United States adopt national codes and standards, products manufactured to these standards can be shipped from one state and installed in another. The use of national standards gives a uniform understanding of what is required in the manufacturing shops. The importance of this can be illustrated by an example of where it did not apply. At one time, a number of states had individual codes applying to residential water heaters requiring the manufacturers to produce a different model for installation in each of the several states. The intent of each state's code was to protect public safety, but the resulting protection was non-uniform. Similar practice on nuclear plants would result in the price of developing practices, obtaining their acceptance, and applying them being paid on an ad hoc individual plant or company basis, rather than being accomplished more efficiently and effectively in a co-operative systematic industry standards group.

The availability of uniform standards throughout an organization makes it easier for a less experienced person to understand the requirements. This better assures that the necessary quality will result.

Utility owners have the final responsibility for ownership, operation and safety of the nuclear power stations built to the codes and standards. The owners therefore have a vital stake in being competently represented on the national committees that develop these codes and standards. The United States practice of using volunteer committees to develop and nurture the standards provides full opportunity for input from owners as well as from government and the manufacturing industry.

CONCLUSION

It is clear that several advantages derive from the systematic industry-wide effort on the development, acceptance, and disciplined application of engineering standards [4].

Standards enhance safety and reliability

We know from substantial experience that nuclear plants can be built with outstanding records for safety and reliability; we have the technology to produce this result.

Standards provide a key means of documenting this technology and making it available to those who need to reproduce these successes. They embody statements of the technology, the engineering procedures and practices, and the specifications which have proven successful in the areas of design, construction, inspection, testing, operation and maintenance in service. Such standards enhance assurance of safety and reliability and help avoid the difficulties and delays that result from not using what we know.

Standards help state requirements and assure they are met

A utility, as purchaser of a nuclear plant, and the various suppliers, constructors, and equipment vendors involved must be able to define, understand and agree upon the requirements to be met. The availability of standards which have been developed and recognized on an industry-wide basis will greatly enhance the ability to do this effectively. Recognized nuclear industry standards provide the needed basis for specifying what the utility requires, for better evaluating competing proposals, and for more effectively defining and enforcing agreements.

Standards facilitate the licensing process

The licensing process requires careful review and attention to many detailed aspects of plant design, construction, and operation. To the extent that accepted engineering standards are available, they will provide a basis for accomplishing these reviews more expeditiously than is possible in their absence. Standards that are developed in consonance with USAEC requirements in this area will help establish an understood basis for meeting these requirements. Standards will thus enhance the effectiveness and speed of the licensing process with advantage both to the industry and the government.

Use of standards has substantial economic advantages

The use of standards cited above – for safety and reliability, for stating and meeting requirements, for facilitating licensing reviews – has substantial economic advantages. Construction delays on a 1000 MW(e) nuclear power plant can cost in excess of US \$50 000 000 per year in interest on construction funds and in increased fuel cost from use of older, less efficient fossil plants.

The nuclear community has accepted the tremendous challenge and responsibility of meeting the unprecedented demands for electricity and other forms of energy essential to health, safety, and welfare. The role of codes and standards in achieving safe, dependable, and economic nuclear power to meet this challenge is abundantly clear.

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LEGAL PROBLEMS OF INTERNATIONAL DANGER PROTECTION AND OF INTERNATIONAL EMERGENCY ASSISTANCE IN THE EVENT OF RADIATION ACCIDENTS

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Abstract—Résumé—Аннотация—Resumen

LEGAL PROBLEMS OF INTERNATIONAL DANGER PROTECTION AND OF INTERNATIONAL EMERGENCY ASSISTANCE IN THE EVENT OF RADIATION ACCIDENTS.

Nuclear accidents involve the risk of widespread international damage because of the special characteristics of radioactivity and the theoretically possible vast extent of the damage. It is therefore not only in the interest of the state in which the accident occurred to check the damage quickly and effectively, but, in the case of major accidents, this is also in the interest of the neighbouring states. With one exception there are no international agreements on effective international measures for danger protection and emergency assistance. International emergency assistance in the event of natural catastrophes and other such major disasters have indeed proved that international solidarity makes it possible for assistance to be provided without special agreements. However, such spontaneous action without a binding legal basis is obviously problematic. Moreover, it must be ensured that the necessary special rescue equipment is at hand. Finally, it is questionable whether the state where the accident occurred will permit foreign relief on its territory without agreeing to special contractual regulations. A refusal to accept foreign relief would lead to complications, particularly if the state's own funds were not sufficient to avert the danger and the risk of international damage constituted a threat.

In this paper, current international treaty law, international custom and the national laws of important individual states are examined as to whether, and to what extent, international emergency assistance and danger prevention across national borders are admissible or even required by law. The paper comes to the conclusion that present-day law is not adequate to secure international catastrophe protection. Principles for a multi-lateral agreement are worked out. In particular, the paper also investigates whether, and how effectively, preventive accident protection can be achieved by multilateral standardization of the licensing prerequisites for nuclear installations and other nuclear activities.

PROBLEMES JURIDIQUES LIES A LA PROTECTION ET A L' ASSISTANCE INTERNATIONALES EN CAS D' ACCIDENTS NUCLEAIRES

En raison des particularités de la radioactivité et de l'immensité des dégâts qui pourraient théoriquement se produire, les accidents nucléaires entraînent le risque de dommages d'une portée internationale. Ainsi l'intérêt d'enquêter les dégâts d'une manière rapide et efficace ne se limite pas au pays dans lequel l'accident s'est produit, mais, dans les cas d'accidents graves, s'étend aux pays voisins. A une exception près il n'y a pas d'accords internationaux fixant des modalités efficaces de protection et d'assistance internationales en cas d'urgence. Il est vrai que la solidarité internationale est telle que des mesures d'assistance sont prises en cas de catastrophes naturelles ou d'autres désastres même à défaut d'accords spéciaux, mais les problèmes que posent de telles actions spontanées qui ne relèvent pas d'obligations juridiques sont évidents. De surcroît, il faut être assuré que les moyens de sauvetage nécessaires sont disponibles et prêts à être utilisés. Enfin on peut douter que l'Etat dans lequel l'accident a lieu autorisera la mise en place d'une assistance étrangère sur son territoire sans réglementation contractuelle. Le refus d'accepter une assistance étrangère conduirait à des complications, notamment si les ressources propres de l'Etat ne suffisent pas à parer au danger et si les dégâts risquent de s'étendre au-delà du territoire national.

Le mémoire examine le droit international conventionnel en vigueur et le droit international coutumier, ainsi que le droit national de certains importants pays, pour déterminer si et dans quelle mesure une assistance internationale et des mesures préventives au-delà des frontières nationales sont permises ou même exigées par le droit. Cet examen démontre que le droit actuel ne suffit pas à assurer sur le plan international la

protection exigée en cas de catastrophe. Les principes d'un accord multilatéral sont étudiés. Le mémoire discute notamment la possibilité et l'efficacité d'une protection préventive qui serait assurée au moyen d'une standardisation multilatérale des conditions d'octroi de licences pour les installations nucléaires et d'autres activités atomiques.

ЮРИДИЧЕСКИЕ ПРОБЛЕМЫ МЕЖДУНАРОДНОГО ПРЕДОТВРАЩЕНИЯ РАДИАЦИОННОЙ ОПАСНОСТИ И МЕЖДУНАРОДНОЙ ЧРЕЗВЫЧАЙНОЙ ПОМОЩИ В СЛУЧАЕ ЯДЕРНЫХ АВАРИЙ.

Ядерные аварии влекут за собой опасность значительного международного ущерба, вследствие специфических характеристик радиоактивности и теоретической возможности быстрого распространения. Поэтому быстрая и эффективная проверка ущерба – в интересах не только тех государств, где произошла авария, но и также, при крупных авариях, – обычно в интересах соседних государств. Однако, за одним исключением, отсутствуют какие-либо международные соглашения по эффективным международным мерам в отношении защиты от опасности и в отношении чрезвычайной помощи. Деятельность по оказанию чрезвычайной помощи в случае стихийных бедствий и других подобных крупных катастроф на деле доказала, что международная солидарность является таковой, что помощь может быть предоставлена без специальных соглашений. Однако такая спонтанная акция, без связующей юридической основы, является, очевидно, проблематичной. Кроме того, необходимо обеспечить постоянное наличие нужного специального спасательного оборудования. Наконец, может возникнуть вопрос, разрешит ли государство, в котором произошла авария, зарубежным странам оказывать помощь на своей территории без согласования специальных договорных регулирующих норм. Отказ принять помощь от зарубежных стран привел бы к осложнениям, особенно в тех случаях, когда собственные ресурсы государства являются недостаточными для предотвращения опасности и когда создается опасность нанесения ущерба в международном масштабе.

В докладе рассматриваются проблемы международного договорного права, международной практики и национального права некоторых крупнейших государств под углом зрения, являются ли вообще, и до какой степени, допустимыми или даже обязательными, в рамках закона, международная чрезвычайная помощь и предотвращение опасности за пределами национальных границ. В докладе делаются выводы, что нынешнее право не является достаточным для обеспечения необходимой помощи при международных катастрофах. Выработаны принципы многостороннего соглашения. В частности, в докладе также рассматривается вопрос о том, может ли быть решена, и насколько эффективно, задача предотвращения аварий путем многосторонней стандартизации требований в отношении выдачи лицензий для строительства ядерных установок, а также для проведения других видов деятельности в области использования ядерной энергии.

PROBLEMAS JURIDICOS RELATIVOS A LA PROTECCION Y A LA AYUDA INTERNACIONALES EN CASO DE ACCIDENTE NUCLEAR.

Por las especiales características de la radiactividad y la gran magnitud que, teóricamente, podrían llegar a tener los daños, los accidentes nucleares entrañan riesgos de alcance internacional. Por esta razón, atajar rápida y eficazmente los daños interesa no sólo al estado donde se ha producido el accidente, sino también, cuando éste reviste grandes proporciones, a los estados vecinos. Con solo una excepción, no existen acuerdos internacionales que establezcan medidas eficaces de protección y ayuda en caso de accidente. Las actividades internacionales de asistencia en caso de catástrofes naturales y otros desastres han demostrado que la solidaridad internacional es tal, que puede prestarse ayuda sin necesidad de acuerdos, pero esta acción espontánea, que no procede de una obligación jurídica, resulta evidentemente problemática. Además, debe asegurarse que el material de salvamento necesario está siempre dispuesto para ser utilizado. Por último, es dudoso que el estado donde ocurrió el accidente aceptará la asistencia extranjera en su territorio sin haber convenido su reglamentación contractual. La negativa a admitir asistencia internacional produciría dificultades, especialmente si los fondos del estado fueran insuficientes para evitar el peligro y existiera la amenaza de que los daños pudieran extenderse a otros países.

En el trabajo se examinan el derecho internacional convencional y el consuetudinario, así como las legislaciones nacionales de algunos estados importantes, para determinar si y en qué medida es admisible o, incluso, impuesta por la Ley la ayuda internacional y la prevención de los daños por encima de las fronteras nacionales. El trabajo llega a la conclusión de que la legislación actual no ofrece las suficientes posibilidades para procurar la protección internacional necesaria en caso de catástrofe. Se estudian los principios para un acuerdo multilateral. Se investiga especialmente si puede conseguirse y la eficacia que tendría una protección preventiva mediante la normalización internacional de los requisitos exigidos para la autorización de instalaciones y otras actividades nucleares.

1. THE PROBLEM

Emergency and catastrophe assistance plans have today largely become an indispensable instrument of well organized human society. The organization of protection from the effects of natural forces such as fire, floods and earthquakes is to the same degree the object of private and state endeavour as is the prevention of harmful effects arising from technical, industrial and other activities. The peaceful use of nuclear energy has produced dangers of a new and special kind. The risks which the use of nuclear energy involves can only be controlled by the use of complicated technical equipment and installations; the expensive techniques require considerable financial means and a high standard of competence on the part of the personnel. In addition, in the event of radiation accidents leading to radioactive pollution of the air and waters, damage affecting several countries may in many cases have to be reckoned with. Under these circumstances, the need for special accident precautions and emergency measures cannot be denied. The peculiarities of radioactivity and its great potential danger seem to stress the importance of organizing an international danger protection system and of taking international measures against radiation accidents. This is first and foremost a technical problem. Plans must be developed on ways and means to meet every conceivable dangerous situation. The necessary personnel and equipment must be at hand and available for use at short notice. In addition, legal problems arise in connection with this task which, if not solved before an accident, may complicate considerably, if not render impossible, the necessary assistance measures.

The legal basis of emergency assistance in the event of radiation accidents on the national level is supposedly clear and unproblematic. Nuclear emergency assistance here is part of already existing danger protection, emergency and catastrophe regulations; these regulations have to be supplemented, if necessary, by special nuclear regulations.

This is different from international protection and emergency assistance measures where the state borders usually form insurmountable obstacles to foreign emergency assistance and can only be overcome by declarations of consent on the part of the states concerned. But even if permitted and desired, the assistance of emergency teams on foreign territory is not without its problems. Apart from the logistical questions of co-ordinating emergency efforts with national measures and overcoming eventual linguistic difficulties, the problem of, for example, which country bears the costs, who settles for accidents to the assisting foreign teams and who settles for damage to third parties caused by these emergency teams has to be settled.

The practical significance of the legal problems mentioned must not, however, be overemphasized. In the case of really serious nuclear catastrophes, e. g. a reactor catastrophe polluting a large area, it will, without doubt, be possible to arrange international emergency assistance. The numerous cases of international relief work when natural catastrophes have occurred prove that it is possible for states to provide mutual assistance quickly, effectively and unbureaucratically, even if not entirely without friction or failure. Nevertheless, it appears necessary, especially because of the intricacy of the matter, to consider the legal bases for international danger protection and emergency assistance and, as far as they

should prove to be insufficient, to secure and to facilitate the accomplishment of the necessary measures of assistance by international agreements. This is especially true for densely populated areas such as Western Europe, where it is of great consequence to check incidents involving damage quickly without time-consuming negotiations beforehand.

The hitherto existing special nuclear emergency assistance agreements are considered here, as well as further treaties of possible relevance in order to see what legal basis is offered for international danger protection in the event of radiation accidents. The possibility of using international customary law is then discussed. In a final section, the problems of possible future regulation by treaty are presented.

2. SPECIAL NUCLEAR DANGER PROTECTION AND EMERGENCY ASSISTANCE AGREEMENTS

2.1. "Nordic Mutual Emergency Assistance Agreement in Connection with Radiation Accidents"

The "Nordic Mutual Emergency Assistance Agreement in Connection with Radiation Accidents", signed by the Scandinavian states and the International Atomic Energy Agency (IAEA), dated October 17, 1963 [1], is to this day the only pertinent multilateral emergency assistance agreement.

The agreement does not contain any explicit regulations concerning the prerequisites for emergency assistance; it regulates only the questions which arise between the contracting parties if emergency assistance is agreed upon and carried through. One might, however, possibly deduce from the preamble of the agreement that the signing parties wished to commit themselves to assisting each other in every case where it is desired ("... desiring to assist each other ..."). If this interpretation is correct — which is, however, very questionable — it would mean that a state seeking help is, in principle, under international law entitled to receive emergency assistance from the contracting parties. In each particular case, however, it remains open to question how far this commitment goes [2]. The obligation to assist in the event of emergency does not mean that a contracting state is obliged to accept the assistance offered in every case. In particular, this agreement does not offer any legal instrument to enable the other states to perform emergency assistance measures against the will of the state concerned.

Articles I, III and IV contain those regulations of the agreement which are of legal and practical importance. According to Article I, the state seeking help assumes full responsibility for effecting the emergency assistance. Throughout the duration of the emergency assistance and while they are on the territory of the state which is seeking help, the foreign assistance teams are subject to the direction and supervision of this state. This secures first of all the co-ordination of all assistance measures. Beyond this, it means that the assisting state does not act within the foreign territory by virtue of its own sovereign power but that it only places manpower and materials at the disposal of the foreign state.

According to Article IV, No. 1, the state seeking assistance bears all "risks and claims" arising at the time of or in connection with the emergency assistance. It has to exempt the state offering assistance and its

personnel from all claims unless the damage has been caused by wilful misconduct and through gross negligence. Damage to body and life of the personnel as well as damage to the equipment has to be covered by the state seeking help. Damage caused within the territory of the assisting state has to be covered by this state itself. The liability regulation of this Article does not apply when damage is covered by liability of the operator of a nuclear installation; in this case the operator is liable for the damage in the state seeking help as well as in the assisting state (Article IV, No. 4). Rights of recourse which may exist under national law remain untouched by the regulation of the Article in the case of claims against the personnel of the assisting state, however, only when the damage has been caused by wilful misconduct or by gross negligence (Article IV, No. 5).

The costs of emergency assistance are so distributed that in principle each party bears those costs which arise within its own territory (Article III). The assisting state, however, can claim restitution of the expenses caused in connection with the emergency assistance so that, as a result, the state seeking help has to bear the costs of the relief work alone.

There is not enough space here to go into further details of the agreement, but it should be noted that, according to Article II of the agreement, the IAEA has to fulfil special functions giving it a central position within the framework of the agreement.

It can be said in conclusion that the 'Emergency Agreement' of the Scandinavian states and the IAEA as it stands offers a practical and reasonable legal framework for emergency assistance in the case of nuclear accidents. Yet it is striking that the more technical questions of organizing the emergency assistance have not been taken into account in the treaty. The assistance has to be organized ad hoc in every single case. It would appear doubtful whether this is suitable. One could, for instance, have thought of appointing a joint committee which would prepare specific plans for emergency assistance. It is questionable whether this has actually happened without explicit stipulations to this effect.

2.2. Agreement between Belgium and France on Radiation Protection at the Centrale nucléaire des Ardennes

An agreement on emergency assistance in the event of radiation accidents was concluded within the limits of the "Convention entre la Belgique et la France sur la protection radiologique concernant les installations de la Centrale nucléaire des Ardennes, et annexes, signée à Paris, le 23 septembre 1966" [3]. The Centrale nucléaire des Ardennes, which is run by the "Société d'énergie nucléaire franco-belge des Ardennes (SENA)" — a joint enterprise in accordance with the Euratom treaty — is situated on French territory near the Belgian border. Because of its application to only one nuclear installation, the regulations of the treaty are less comprehensive than those of the Scandinavian Emergency Agreement. On the other hand, this treaty contains just those regulations which are lacking in the other agreement; that is, regulations on the organization of emergency assistance measures.

Article 4 of the treaty settles the organizational principle of the emergency assistance: "En cas d'accident, les Parties contractantes désireuses

de s'aider dans toute la mesure du possible, placent les moyens qu'elles fournissent sous une autorité unique qui assurera la direction générale des secours et de l'intervention". It is comparable to the Scandinavian agreement in as much as only one authority is declared competent, and the foreign assistant teams have to be put under its control. The necessary organizational details are regulated in an annex to the agreement. According to Article 5, the contracting parties commit themselves to confine to a minimum the formalities for the persons, vehicles and materials when crossing the border. A mixed commission is competent for arranging radiation protection and the particular problems which accrue from the agreement (Article 6).

Annex III provides the following regulations for mutual emergency assistance: each contracting party has a direct claim to the competent assistance services in the event of a serious accident; the contracting parties organize groups in their border zones and inform each other of their available equipment and quantity of vehicles; the competent national authorities draw up a concerted plan for emergency assistance which stipulates the organizational details. It should be mentioned that this agreement neither provides for a compensation for costs nor for a liability regulation.

If one looks for the reasons behind these striking differences between the Belgian-French and the Scandinavian emergency assistance agreements, one must seek them predominantly in the different geographical scope of the treaties. The Belgian-French treaty is limited to one single nuclear installation which, in addition, is situated so closely to the border of the two contracting parties that both countries would necessarily be affected in the case of serious accidents. Moreover, the associates of the SENA, which runs the nuclear installation, are the Belgian stock corporation "Centre et Sud" and the French "Electricité de France". Co-operation in the event of damage is obviously called for here. Cases of damage are, in spite of the dividing border, internal events for which there must be a technical alarm plan. Legal questions play a secondary role here.

The situation is somewhat different in the Nordic multilateral agreement, in which four states and one international organization are the contracting parties. In the Scandinavian region, there are many reactors and other installations where radioactivity is being used. In this case, many different assistance plans had to be developed. Co-ordination is more difficult because there are four borders and as many different legal systems. Because of this multiplicity, more legal questions naturally come to be discussed than in the case of the Belgian-French co-operation. As a consequence of the necessarily greater unwieldiness of multi-national co-operation, the effectiveness of emergency assistance suffers because of possible lack in specific assistance plans. The first deduction can already be derived from this evaluation: with an increasing number of participants in an emergency assistance agreement, the legal apparatus grows. In these cases an unambiguous legal basis for the assistance and the problems connected with it is necessary. The importance of legal problems recedes if the emergency assistance is limited to two or three states at most, especially when the objects of the assistance are few in number and perhaps even close to the border.

2.3. The Radiation Emergency Warning System of the OEEC/OECD

"The Supervision and Emergency Warning System in Cases of Increase in Environmental Radioactivity", constituted on the basis of a decision of the council of the OEEC on 7th July 1961 [4], is different from the treaties mentioned before. Here it is not a question of guaranteeing emergency assistance after an accident has occurred, but an opportune notification of the increase in environmental radioactivity is intended to render possible preventive measures for protection in the event of danger.

The organizational structure of the radiation warning system is simple: each state determines a competent authority which is alone competent and which conveys and can receive information at any time. The following events must be announced: nuclear incidents or massive increases of radioactivity which endanger the health of a population in an entire region ("Emergency Cases"); massive increases of radioactivity which are of undefined origin or which originate from a smaller incident and demand no local radiation protection measures ("Pre-Emergency Cases"); unusual increases of radioactivity of scientific interest ("Cases of Scientific Interest"). The results of measurements are to be sent to the competent centres in the neighbouring states and to the European Nuclear Energy Agency (ENEA) of the OECD, whereby different transmission routes are prescribed depending on the importance of the announcement.

It is obvious that the warning system of the OECD can be no alternative to the agreements described above. It would nevertheless be a meaningful supplement to treaties which do not consider the preventive aspect of danger protection. One could also conclude that a combination of the warning system of the OECD with the Nordic and Belgian-French emergency assistance agreements already comes very close to a conceivable ideal of an international danger protection and emergency assistance system.

3. OTHER INTERNATIONAL TREATIES AND INTERNATIONAL CUSTOMARY LAW

3.1. Treaty Establishing the European Atomic Energy Community

The system of health protection set up in the European Atomic Energy Community according to Article 30 et seq., Euratom Treaty [5] implies that organizational and other precautionary measures for preventing and checking radiation accidents have been agreed upon. Nevertheless, special regulations are lacking. This lack does not seem to be terribly important when one considers that the law of radiation protection in the countries of the community is almost completely harmonized according to Article 30 et seq., so that far-reaching preventive protection against radiation danger has already been achieved. Furthermore, according to Articles 35 and 36 of the treaty, a radioactivity surveillance and warning system exists by which a dangerous increase of the environmental radioactivity can quickly be recognized. Finally, in especially urgent cases, with the directives based on Article 38, paragraph 2 of the treaty, the Commission can arrange the necessary measures. All this doubtless circumscribes a legal situation which renders possible international danger protection and emergency assistance.

Nevertheless, it should be queried whether, without specific plans and other organizational preparation, an optimal assistance system between the contracting countries can be implemented. The Commission has recognized this problem: "The possibility of intervention in radioactive accidents assumes the solution of a number of administrative and legal questions before the specific technical measures can commence" [6]. Corresponding deliberations have been made, as shown in the answer to an inquiry by Deputy Santero in the European Parliament on 26 November 1965 [7]. These deliberations have not, however, brought any results. Catastrophe plans have been developed solely for the installations of the joint nuclear research centres in Ispra, Karlsruhe, Petten and Geel, but no details have been published. For national research centres, emergency assistance plans have also been established, apparently in co-operation with the governments concerned [8]. Apart from this, no multi-national danger protection and emergency assistance system exists between the Member States of the European Atomic Energy Community.

3.2. International Atomic Energy Agency (IAEA)

Although the IAEA is, according to its Statute, the most qualified international organization to build up a system of danger protection and emergency assistance in the event of radiation accidents, one cannot expect too much, since its legal possibilities are limited. To be sure, one of its tasks is to assist its member states in cases of accident, if required; it can also make recommendations on the most suitable kind of assistance [9]. It cannot, however, issue obligatory regulations.

The IAEA started dealing with the problems of international emergency assistance in radiation accidents quite early on. The first discussions took place in the spring of 1958 [10] and resulted in the Nordic Agreement of 1963. In 1964, the Eighth General Conference of the IAEA passed a resolution in which the Board of Governors was formally charged with the task of taking the measures necessary to promote the conclusion of emergency assistance agreements [11]. The then prevailing optimism about being able to incorporate the principal features of the Nordic Agreement into a universal agreement faded away however. The deliberations dragged on until 1967 and were then suspended. Most of the states concerned consented to render assistance in the event of accidents at any time, but were not willing to submit beforehand to the obligations of an agreement. The main points of the discussion were the regulation of the questions of liability, the question of whether the IAEA should or should not be a party to the agreement, the concession of privileges and exemptions and, finally, the question of whether a universal, a regional or a bilateral agreement would be more suitable.

Apart from this, the delegates of the participating states emphasized over and over again the significance of the matter, and they pointed out that the work of the IAEA would not have been in vain even if no agreement were reached.

The work was not in vain, since four model agreements were developed which can be placed at the disposal of the Member States for the conclusion of future treaties. The draft of a multilateral agreement of which the IAEA is a party, a multilateral agreement of which the IAEA is not a party, a

bilateral agreement between states, and a bilateral agreement between one state and the IAEA all exist, although unpublished and unfortunately inaccessible to general discussion. The model drafts resemble very much in structure and content the Nordic assistance agreement. Two decisive points are, however, different: regulations on liability are lacking and there is no claim to emergency assistance; to quote: "Neither the Agency nor any state is obliged to request or accept, offer or provide assistance by reason merely of being a party to this Agreement". In this respect, these drafts are a step backwards in comparison with the Nordic agreement. The ten years of deliberation of the IAEA and their final outcome show very distinctly how difficult it is to accomplish effective regulations on the international level even if there is basic unanimity as regards the necessity of these regulations.

Attention must be drawn to two treatises of the IAEA which are at least as significant as the model agreements for the application of emergency measures. One of them is the manual "Planning for the Handling of Radiation Accidents" [12], worked out in co-operation with the ILO, FAO and WHO, which offers fundamental directions for the handling of radiation accidents. The other is the document "Mutual Emergency Assistance for Radiation Accidents" [13], which was prepared jointly with WHO and FAO and in which the composition, equipment, efficiency and capability of national radiation rescue teams are treated.

3.3. Other legal bases and international customary law

It would be fascinating to investigate further international treaties or statutes of other international organizations to and a legal basis for international danger protection and emergency assistance in radiation accidents. One would certainly be able to discover corresponding competence in the statutes of WHO or ILO — perhaps not of such comprehensive nature as in the Statute of the IAEA, but confined to a specific range of functions. Likewise, the United Nations Organization has the statutory right to deal with this complex of problems and to issue pertinent recommendations. At present, however, all these competences are at the most of theoretical interest since no activities of these organizations could so far be observed in this field except for the co-operation of the ILO, FAO and WHO in IAEA projects. Finally, it must be pointed out that legal bases for emergency assistance could probably be found in treaties of friendship and co-operation and in other bilateral and multi-lateral agreements.

In recapitulating this survey of the international law of treaties, it can be said that there are not enough treaties regulating the problems of international emergency assistance in radiation accidents. The model agreements of the IAEA exist, but it is open to doubt whether in the event of damage, there will actually be enough time to put into force these treaties between the parties concerned. Furthermore, these drafts are kept in the IAEA and marked "restricted", so that any encouraging public discussion is impossible. Yet it remains to be investigated if the gaps in treaty law can be covered by statutes of the international customary law. Or to express it in a different way: Is the international customary law sufficient to render international emergency assistance possible so that a specific treaty law is no longer necessary?

In principle, the answer to this question must be in the negative. The necessary crossing of state borders in the case of international emergency assistance generally requires the explicit consent of the states concerned and thus a treaty. It is, however, conceivable that locally limited customary law could exist to the effect that neighbouring communities can cross the borders and help each other without formalities when catastrophes and emergencies occur. Such laws, however, should not play any practical role in the problems which are being discussed here. In general, no statute of international customary law can be found which obliges any state to assist another state on request or which authorizes anyone to force help upon another state against its will. Radiation accidents within the territory of one state are, in principle, not the business of another state.

The legal situation has to be judged differently when a radiation accident has effects reaching beyond national borders. In this case the accident becomes relevant in respect to international law. If the state which causes the damage refuses the offer of foreign assistance although it cannot control the effects of an accident with its own means, one must qualify its conduct – in accordance with the principles developed in the 'Trail Smelter' Case [14] – as an unlawful act, a fact which makes damage settlement obligatory. In the case of a state's own incapability and the danger of damage with effects reaching beyond national borders, an international legal obligation exists to solicit and accept foreign assistance. This, however, does not guarantee the right to third parties to carry through by force assistance measures in foreign territory¹.

On the other hand, it will hardly be possible to oblige a state to assist another state merely on the basis of general statutes of international law. Without doubt, it is to be condemned if a state calmly watches a reactor catastrophe in another country without rendering effective assistance even if it has been asked to do so. Unless particular circumstances present themselves, however, it might be difficult, without seriously offending against international law, to draw up an international legal obligation to render emergency assistance².

4. PROBLEMS OF FUTURE REGULATION BY AGREEMENT

The paper has shown that the current international law of treaties and the current international customary law do not offer a legal basis which is sufficiently reliable for an international danger protection and emergency assistance system. This legal situation is without doubt unsatisfactory, even if one assumes that the technical problems of emergency assistance are more important than the legal ones. Although many states are aware of this situation, it is difficult to believe in a real chance for the creation of comprehensive international regulations. The failure of the IAEA endeavours is not yet forgotten; the objections which then existed would probably be raised again today.

¹ In the event of a threat to the existence of a state through reactor catastrophes abroad, the right to forcible assistance from the point of view of necessity (right of self-preservation) might be considered.

² An obligation of this kind could possibly be derived from previous action: the highly developed state A delivers a reactor to developing country B, knowing that the latter does not have trained reactor personnel.

To find a new approach, a considerable impulse from outside would be necessary. A nuclear accident would certainly be most effective in drawing attention to the lack of legal regulations. The 'Torrey Canyon' Accident has proved that, after a catastrophe, states are more disposed to accept new legal obligations [15]. Naturally we all hope that such an 'incentive' accident will never happen. It would be less spectacular and incisive if the United Nations attended to this complex of problems. A UN resolution would certainly carry so much weight that deliberations on this matter would be started anew. At present it seems to be conceivable that the UN might take up these problems, since environmental protection is of wide interest today. Moreover, only recently the Economic and Social Council of the UN dealt with the general problems of 'Emergency Assistance' in connection with the earthquake catastrophe in Peru in 1970 [16]. If it can be made clear that nuclear emergency assistance is a section of 'Environmental Protection' as well as of the entire complex of 'International Emergency Assistance', there should be a real chance for these questions to be considered again.

As an international emergency assistance agreement, a multilateral and, if possible, universal agreement would be ideal. That universally planned emergency assistance agreements are not impossible was proved recently in a different field by the international agreement on the rescue of astronauts [17]. On the other hand, the degree of legal complexity increases with the number of participating states. The legal structure of the agreement becomes more unwieldy and its practical efficacy decreases. If it were at all possible to accomplish a treaty of universal scope, a limitation to a general framework of regulations would be advisable, which might be complemented by regional supplementary agreements. Bilateral or locally limited multilateral treaties are supposedly of greater practical significance because they form the best basis for organizing an emergency assistance system which would fulfil the actual demands and, over and above regulation of the legal bases, stipulate the technical points of emergency assistance. With regard to the efficacy, geographically limited agreements, or at least agreements with a limited number of participants, should be preferred³.

With the example of the Nordic emergency agreement in mind, it might also be suitable to make the IAEA a party to such treaties⁴. The deliberations within the IAEA on 'Emergency Assistance' have shown, however, that the participation of the IAEA is regarded by many states as problematic. It was denied that the IAEA could be a party to the international emergency assistance agreements in its own right. Whether or not this is correct will not be discussed here, since it is not absolutely necessary to make the IAEA a contracting party. In accordance with its statute, the IAEA is entitled to assist its Member States if required, and its technical potential thus available even without an agreement⁵.

³ This may be different for countries which, because of their stage of development, are not always able to help each other. Here the participation of adequately equipped states is always desirable, even if they are not geographically neighbouring states.

⁴ This could apply to other international organizations as well.

⁵ The questions of liability and compensation for costs would, of course, also require special regulations in these cases.

The question of whether a legal claim to emergency assistance should be justified with an assistance agreement is above all a political one. The answer depends, among other things, on the general relations of the potential contracting states: states on friendly terms will be more inclined to assume an obligation to give or receive emergency assistance. The question of establishing an explicit claim to emergency assistance should, however, be treated with some reserve, since it might become an obstacle to an emergency assistance agreement. The efficacy of an eventual agreement should not depend on whether or not such a claim exists. Should it be decided, however, to include in the treaty an international legal assistance claim, it would be advisable to give an exact definition of the cases of emergency assistance and to stipulate the kinds of radiation accident which would justify the claim.

The contents of the agreement can be divided into two parts: one regulating the technical and organizational problems and the other regulating the legal questions connected with these problems. This would be approximately a combination of the Belgian-French and the Nordic emergency agreements. It does not, however, seem to be advisable to include in the treaty specific emergency assistance plans. This would not only complicate the treaty but would fix the emergency plans to such an extent that adaptation to new circumstances would be a slow and unwieldy process. The task of preparing emergency plans could instead be entrusted to mixed commissions of experts [18].

Further questions to be regulated in the agreement concern the legal position of the assistance teams in a foreign country and, especially, the questions of liability and costs.

As to the legal status of the foreign assistance teams, a wide scale of possibilities is offered between a quasi diplomatic status including privileges and exemptions on the one hand and treatment as normal foreigners on the other hand. The assisting state would naturally prefer a legal position for its assistance teams which would be as favourable as possible. This is especially true when personnel are sent who exercise public authority in their own state, e. g. radiation protection teams of national fire-brigades or even people belonging to the army. In such cases it will be almost impossible to determine the legal situation of the assistance teams on a fixed basis, and it will be necessary for the contracting parties to negotiate in each particular case.

The costs of assistance should generally be borne by the state to which assistance is rendered. A regulation in this regard should be adequate and attainable. As regards liability, there must be differentiation between damage caused by the assistance team, and damage suffered by it. In the first case, a comprehensive exemption of the assistance team from personal liability claims must be aimed at, with the exception of damage which has been caused with intent or by gross negligence. The claims against the assisting state should also be limited to damage caused with intent or by gross negligence. If the assistance team suffers damage in connection with the emergency assistance, an immediate settlement of claims has to be arranged, if possible, by the state being assisted. A reference to eventual claims against the responsible operator of a nuclear installation [19] would not be suitable, since the assisting state and its assistance team cannot be expected to engage in a lawsuit against the operator.

Finally, consideration will have to be given to the extent to which the purpose of the treaty can be furthered by preventive clauses. A radiation supervision system on the pattern set by OECD or Euratom would give timely warning of eventual sources of danger. Provisions to encourage unification of licensing and supervision laws in the contracting states would be welcome. When the standards of safety in the contracting states are homogeneous in content, not only will the risks of an accident decrease but assistance teams will then move on familiar ground, so to speak, at foreign installations, so that emergency assistance will be less complicated and, therefore, more effective. It is doubtful, however, whether such far-reaching provisions in danger protection and emergency assistance agreements are desirable and practicable, they might impede the realization of a treaty. The unification of the preventive regulations must be carried through on a wider basis, and emergency assistance agreements should therefore be limited to the scope of direct emergency assistance. Only then will there be some hope of achieving treaty regulations and of closing the gap in the system of the international law on radiation protection.

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SHALL NUCLEAR SHIPS BE STRANDED FOR LACK OF RELEVANT LEGAL PROVISIONS?

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Abstract—Résumé—Аннотация—Resumen

SHALL NUCLEAR SHIPS BE STRANDED FOR LACK OF RELEVANT LEGAL PROVISIONS?

In no field of modern technology has the legal basis been so well prepared internationally as for nuclear ships. Already in 1960, when only one civilian nuclear ship, the USSR 'Lenin', was in operation, a special chapter on nuclear ships was included in the London Convention on Safety of Life at Sea. In May 1962, a Diplomatic Conference at Brussels agreed on a Convention on the Liability of Operators of Nuclear Ships. Before the 'Savannah' went on international voyages, the United States had concluded, or were negotiating, bilateral agreements with more than a dozen Governments. Thus one could have assumed that the third civilian nuclear ship, 'Otto Hahn', would not encounter major legal difficulties. The contrary proved to be the case. Although the German Government when proposing bilateral agreements on the use of territorial waters and ports to Norway, the Netherlands and some other countries more or less followed the pattern set by the 'Savannah' Agreements, negotiations dragged on, not so much for technical, but mainly for legal and procedural reasons. However, several West-African States and Iran permitted the 'Otto Hahn' to enter their ports after a simple exchange of diplomatic notes. Thus, the ship was able to begin its commercial operations after having made extensive and successful trial runs.

The German Government is now considering ratification of the Brussels Convention on the Liability of Operators of Nuclear Ships. It is hoped that other countries will follow. When both partners to future negotiations are parties to the London SOLAS and the Brussels Convention, things will be much easier and it may be sufficient to agree on necessary details by an exchange of letters. It might be advisable, however, to re-examine the proposals already made at international conferences and aiming at some kind of international procedure to make nuclear ships more readily acceptable in foreign ports. The IAEA and IMCO will have a role to play in this respect, together with those countries which already have experience in the operation of civilian nuclear ships or which intend to operate such ships in the foreseeable future.

LE MANQUE DE DISPOSITIONS LEGALES DOIT-IL ENTRAINER LA PARALYSIE DES NAVIRES NUCLEAIRES?

Dans aucun domaine de la technologie moderne les bases légales n'ont été préparées, sur le plan international, avec autant de soin que dans celui des navires nucléaires. Dès 1960, alors qu'un seul navire nucléaire civil, le bâtiment soviétique Lénine, était en service, on a intégré dans la Convention de Londres relative à la sécurité des navires un chapitre spécial sur les navires nucléaires. En mai 1962, une conférence diplomatique tenue à Bruxelles s'est mise d'accord sur une convention relative à la responsabilité des exploitants de navires nucléaires. Avant que le navire Savannah entreprit des voyages internationaux, les Etats-Unis avaient conclu des accords bilatéraux avec plus d'une douzaine de gouvernements ou avaient engagé des négociations sur ces accords. On aurait donc pu supposer que le troisième navire nucléaire civil, le Otto Hahn, ne se heurterait pas à des difficultés juridiques majeures. Or, c'est le contraire qui s'est produit. Bien que les propositions présentées par le Gouvernement allemand à la Norvège, aux Pays-Bas et à un certain nombre d'autres pays en vue de la conclusion de conventions bilatérales relatives à l'utilisation des eaux territoriales et des ports suivissent plus ou moins l'exemple de l'accord pour le navire Savannah, les négociations traînaient en longueur, moins pour des raisons techniques que, essentiellement, pour des raisons juridiques et de procédure. D'autre part, plusieurs Etats ouest-africains ainsi que l'Iran permettaient au navire Otto Hahn l'entrée dans leurs ports sur la base d'un simple échange de notes diplomatiques. C'est ainsi que le navire, après de longs voyages d'essai couronnés de succès, a pu être exploité sur une base commerciale.

Le Gouvernement allemand envisage actuellement de ratifier la Convention de Bruxelles relative à la responsabilité des exploitants de navires nucléaires. On espère que d'autres Etats suivront. Si les deux partenaires de négociations futures sont parties à la Convention de Londres relative à la sécurité de navires et à la Convention de Bruxelles, la tâche en sera considérablement facilitée et un échange de lettres suffira pour se mettre d'accord sur les détails nécessaires. Pourtant, il sera probablement opportun de soumettre à un nouvel examen les propositions faites à l'occasion de conférences internationales en vue d'introduire

une procédure internationale qui faciliterait l'entrée de navires nucléaires dans des ports étrangers. L'AIEA et l'OMCI y auront un rôle à jouer ainsi que les pays qui ont déjà de l'expérience dans l'exploitation de navires nucléaires civils ou qui envisagent d'exploiter de tels navires dans un proche avenir.

ОСТАНУТСЯ ЛИ ЯДЕРНЫЕ СУДА СИДЕТЬ НА МЕЛИ ИЗ-ЗА НЕДОСТАТКА СООТВЕТСТВУЮЩИХ ПОЛОЖЕНИЙ?

Ни в одной области современной техники юридические основы не разработаны так хорошо в международном масштабе, как в области применения ядерных судов. Уже в 1960 году, когда в эксплуатации находилось лишь одно гражданское ядерное судно, а именно атомход "Ленин" в СССР, специальная глава о ядерных судах была включена в Лондонскую конвенцию о сохранении жизни на море. В мае 1962 года дипломатическая Конференция в Брюсселе одобрила Конвенцию об ответственности операторов ядерных судов. Прежде чем судно "Саванна" отправилось в международное плавание, Соединенные Штаты Америки заключили или вели переговоры о заключении двусторонних соглашений более чем с десятью правительствами. Таким образом, можно было бы полагать, что третье гражданское ядерное судно — "Отто Хан" не столкнется со значительными юридическими трудностями. Однако произошло обратное. Хотя правительство ФРГ, предлагая заключить двусторонние соглашения об использовании территориальных вод и портов Норвегии, Нидерландов и ряда других стран, более или менее придерживалось модели, характерной для соглашений о "Саванне", переговоры затянулись не столько в силу технических причин, сколько в силу причин, главным образом, юридического и процедурного характера. Однако ряд западно-африканских государств и Иран, в результате простого обмена дипломатическими нотами, разрешили судну "Отто Хан" заходить в свои порты. Таким образом, это судно, после прохождения интенсивных и успешных испытаний готово к коммерческой эксплуатации.

Правительство ФРГ рассматривает в настоящее время вопрос о ратификации Брюссельской конвенции об ответственности операторов ядерных судов. Оно надеется, что и другие страны последуют его примеру. Если оба партнера по будущим переговорам будут являться участниками Лондонской конвенции о сохранении жизни на море и Брюссельской конвенции, то решение многих вопросов окажется значительно проще и будет достаточно, по-видимому, лишь согласовать необходимые детали путем обмена письмами. Однако представляется целесообразным еще раз изучить предложения, уже высказанные на международных конференциях и направленные на выработку своего рода международных процедур, которые облегчили бы заход ядерных судов в иностранные порты. Определенную роль в этом отношении должны будут сыграть МАГАТЭ и ИМКО вместе с теми странами, которые уже имеют опыт эксплуатации гражданских ядерных судов или которые намерены приступить к эксплуатации таких судов в недалеком будущем.

¿SE INMOVILIZARÁN LOS BUQUES NUCLEARES POR LA FALTA DE NORMAS JURÍDICAS ADECUADAS?

No existe campo en la moderna tecnología en que la base jurídica se haya elaborado tan cuidadosamente a escala internacional como en el de los buques nucleares. Ya en 1960, cuando sólo estaba en explotación un buque nuclear civil, el soviético «Lenin», se incluyó un capítulo especial en el Convenio de Londres sobre la Seguridad de la Vida Humana en el Mar. En mayo de 1962, una Conferencia Diplomática, celebrada en Bruselas, aprobó un Convenio sobre Responsabilidad de los Explotadores de Buques Nucleares. Antes de que el «Savannah» emprendiera viajes internacionales, los Estados Unidos habían concluido o estaban negociando acuerdos bilaterales con más de una docena de Gobiernos. Por tanto, hubiera podido presumirse que el tercer buque nuclear civil, el «Otto Hahn», no encontraría grandes dificultades jurídicas. Pero ocurrió lo contrario. Aunque el Gobierno alemán, al proponer acuerdos bilaterales a Noruega, Países Bajos y algunos otros países para la utilización de sus puertos y aguas territoriales, siguió aproximadamente el mismo sistema que se establece en los «Acuerdos del Savannah», las negociaciones se prolongaron interminablemente por razones, no tanto técnicas, sino especialmente jurídicas y de procedimiento. Sin embargo, varios Estados del África Occidental e Irán permitieron la entrada en sus puertos del «Otto Hahn» tras un simple cambio de notas diplomáticas. Así el buque, después de haber realizado con éxito largos viajes de pruebas, pudo empezar a ser explotado comercialmente.

En la actualidad el Gobierno alemán considera la posibilidad de ratificar el Convenio de Bruselas sobre la responsabilidad de Explotadores de Buques Nucleares. Se espera que otros países lo hagan a continuación. Cuando, en el futuro, ambos Estados sean partes del Convenio de Londres SOLAS y del Convenio de Bruselas las negociaciones serán mucho más sencillas y un cambio de cartas bastará para acordar los detalles necesarios. Sin embargo, podría ser conveniente volver a estudiar las propuestas presentadas en anteriores conferencias internacionales para establecer un procedimiento internacional que facilite la entrada de buques nucleares en puertos extranjeros. En esta tarea tendrán un papel que desempeñar el OIEA y la OMCI, junto con aquellos países que ya tienen experiencia en la explotación de buques nucleares civiles o que se proponen explotar tales buques en un futuro próximo.

1. INTRODUCTION

The pointed question raised in the title of this paper should, of course, be answered by an emphatic NO. But to people who, like the author, are attempting to open the sea-lanes of the world and the more important foreign ports to a specific nuclear ship, the answer may not appear so obvious. In 1969 an even larger public became aware of the problems faced in this respect by nuclear ships when they read articles that appeared in European newspapers under headlines such as "OTTO HAHN is Undesirable", "Nuclear Ship before Closed Ports", or "Flying Dutchman with Nuclear Propulsion" [1]. How could that be after international conventions governing the safety aspects of nuclear ships and compensation for nuclear damage caused by such ships had been on the books since 1960 and 1962, respectively? Were these conventions inadequate or, if that were the case, were there no alternative solutions to the various problems? In this paper I shall attempt to answer these questions.

2. INTERNATIONAL CONVENTIONS

2.1. International Convention on the Safety of Life at Sea

The spectacularly successful and well-publicized achievements of American naval ships equipped with nuclear propulsion, such as the transit from the Atlantic to the Pacific Ocean via the North Pole by the US submarine NAUTILUS, the submerged circumnavigation of the globe by the TRITON, or the operations of the aircraft-carrier ENTERPRISE, which is equipped with eight reactors, may have been responsible for the high expectations also raised with respect to nuclear-powered merchant ships. Interested circles began studying the legal problems involved in the admission and operation of such ships. When preparations were made by the International Maritime Consultative Organization (IMCO) for the International Conference on the Safety of Life at Sea (SOLAS), held in London from 17 May to 17 June 1960, the establishment of a Committee on Nuclear Ships was provided for. The Committee had before it proposals regarding safety aspects of nuclear ships submitted by the Governments of Australia, France, Italy, Japan, Norway, the United Kingdom, the United States of America, and the USSR [2].

Among the substantive questions discussed in the Committee were two that are of prime importance in the context of this paper:

- (a) Should a regulation that deals with the acceptance of nuclear ships in foreign ports be included in the convention, and, if so,
- (b) Should it be stated that a Safety Report for the nuclear ship should be submitted to the host government for its evaluation before the ship calls at the foreign port?

Both questions were decided in the affirmative.

In the end, in the light of the report and the proposals of the Committee on Nuclear Ships, it was decided in the Conference to include a new Chapter VIII in the 'International Convention for the Safety of Life at Sea 1960', and to give 'Recommendations Applicable to Nuclear Ships' in Annex C to the Final Act of the Conference [2a].

It is thus provided that all regulations of the Convention applying to conventional ships are also valid for nuclear ships (Chapter VIII, Reg. 2). A nuclear ship shall not be exempted from compliance with any regulations of the Convention, as is exceptionally possible for other ships (Reg. 3). The design, construction, and standards of inspection and assembly of the reactor installation are subject to the approval of the Administration (Reg. 4), which means the Government (Chapter I, Reg. 2(b)). The Administration shall take measures to ensure that there are no unreasonable radiation or other nuclear hazards, at sea or in port, to the crew, passengers, or public, or to the water-ways or food- or water-resources (Reg. 6). Very important is the 'Safety Assessment' which must be prepared to permit evaluation of the nuclear power plant and the safety of the ship to ensure that there are no unreasonable radiation or other hazards. The Safety Assessment shall be approved by the Administration and always be kept up-to-date (Reg. 7(a)).

If a nuclear ship intends to visit another country, the Safety Assessment shall be made available sufficiently in advance to the host Government so that they may evaluate the safety of the ship (Reg. 7(b)). Before entering the ports and in the ports of the host country, the nuclear ship shall be subject to special control directed towards verifying that there is on board a valid Nuclear Ship Safety Certificate and that there are no unreasonable radiation or other hazards (Reg. 11). In the event of any accident likely to lead to an environmental hazard the master of the nuclear ship shall immediately inform his own Administration, but also the competent Governmental Authority of the country in whose waters the ship may be or whose waters the ship is approaching in a damaged condition (Reg. 12).

The twelve Regulations of Chapter VIII, as well as the Recommendations in Annex C to the Final Act, were drafted in such a way as to be strict enough to ensure the utmost technical safety and protection of the crew, passengers, or public, and the water-ways or food- or water-resources, but at the same time being broad enough not to hinder technical progress. It appears that these aims were achieved. In the years since the closing of the IMCO Conference, the International Convention on the Safety of Life at Sea 1960 has been ratified by, and come into force for, most seafaring nations. Thus it provides an international legal basis for the safe construction and operation of nuclear ships.

2.2. International Convention on the Liability of Operators of Nuclear Ships

The first large-scale international discussion of problems of liability for nuclear damage caused by nuclear ships took place in Rijeka in 1959 during the 24th Conference organized by the International Maritime Committee. The Committee elaborated a draft convention, certain recommendations and a report [3].

At the invitation of the Director General of the International Atomic Energy Agency, a Panel of experts met in Vienna from 7 to 12 March and from 1 to 7 August 1960. The Panel was presided over by Mr. Albert Lilar, who had also chaired the Rijeka IMC Conference. The experts, specialists in the field of international or maritime law, had before them the Rijeka Draft Convention and other documents prepared by the IAEA Secretariat. They also considered a report prepared by a group of scientific advisers who met to answer some of the questions raised by the Panel during the March session. The Panel submitted an elaborate report to the Director General of the IAEA on the major legal problems in the field of third-party

liability for nuclear ships. This report served as a basis for a Draft Convention which was prepared by the IAEA Secretariat. The report of the Panel was distributed to Governments as a Working Paper for the Diplomatic Conference which was held in Brussels from 17 to 29 April 1961 at the invitation of the Government of Belgium [4].

The time of only two weeks allotted to this 11th Diplomatic Conference on Maritime Law proved to be too short and the legal issues involved were too new and too complicated for easy results. It was therefore not possible in the time available to agree on a Convention that would have been acceptable to a significant number of delegations. Therefore it was agreed to reconvene the Diplomatic Conference in the following year [4a].

When the delegations met again in Brussels in May 1962 for the second phase of their deliberations, it turned out that they had used the intermission very well by studying and becoming familiar with the legal problems of liability for nuclear damage. Thus the Conference succeeded in elaborating a full International Convention on the Liability of Operators of Nuclear Ships. This was opened for signature on 25 May 1962 [5].

The Convention follows the pattern established by the Liability Conventions for land-based nuclear installations in its basic principles, namely:

(a) objective and sole liability of the operator for nuclear damage caused by a nuclear incident involving the nuclear fuel of, or radioactive products or waste produced in, his ship (Article II);

(b) limitation of the operator's liability in amount and time (Articles III, V);

(c) obligation of the operator to cover his liability by insurance or other financial security (Article III, 2); and

(d) obligation of the 'Licensing State', that is the State under whose flag the ship is authorized to be operated (Article I, 2), to ensure payment of claims for compensation for nuclear damage within the prescribed limit to the extent that the insurance or other financial security is inadequate (Article III, 2).

The limit of liability was set at 1500 million gold francs (roughly US \$100 million). The sums provided by the insurance, other financial security or by State indemnification shall be exclusively available for compensation due under the Convention. An action for compensation can be brought either before the courts of the Licensing State or before the courts of the contracting State or States in whose territory nuclear damage has been sustained. A final judgement entered by a court having jurisdiction under the Convention shall be recognized in the territory of any other contracting State, except where the judgement was obtained by fraud or where the operator was not given a fair opportunity to present his case. This execution clause together with the obligation of the operator to cover his liability by insurance or other financial security, and the obligation of the Licensing State to ensure payment of claims for compensation are the outstanding advantages of this Convention [6].

Of the 55 delegations that attended the Diplomatic Conference in Brussels, only 15 have so far signed the Convention. The delegations of the United States of America and of the Soviet Union, which were at that time the only states to operate nuclear ships – the SAVANNAH and the LENIN, had declared in the course of the debates that they would neither sign nor ratify the Convention because it was also applicable to warships.

Under Article XXIV the Convention enters into force when it is ratified by at least one Licensing State and one other state. The United States and USSR declarations meant therefore that for the time being the Convention could not come into force. This was of no concern to the Soviet Union, because their icebreaker LENIN was operating only in Soviet waters or Arctic waters outside foreign territory. However, for the United States of America, who wanted to demonstrate the practicability of nuclear propulsion for merchant ships by sending the SAVANNAH to numerous foreign countries and ports, this meant that they had to resort to other kinds of legal instruments.

3. SAVANNAH AGREEMENTS

In the early stages of planning for the SAVANNAH as the first nuclear merchant ship it had been stated that one of the purposes would also be to develop nuclear law in this field. As early as 1960, even before experience with the operation of the SAVANNAH in American waters and ports had been gained, the United States Government sent draft bilateral agreements to the governments of countries to which they intended to send the ship. These drafts were relatively brief. They were based mainly on the SOLAS Convention and contained some additional provisions that host governments might like to see included. As to liability for nuclear damage, national law of the host state applied and the United States Atomic Energy Commission undertook to indemnify the operator of the ship, Maritime Administration, for an amount of up to US \$500 million. For this purpose, the provisions of the so-called Price-Anderson Act had been amended so as to apply also to the SAVANNAH.

In their final form, the bilateral agreements that were concluded by the United States Government with more than a dozen other governments differ in various details according to requirements of national law or local conditions. Generally, they contain arrangements on questions of liability, radiation protection, safety assessment and, as appropriate, national licences [7].

The SAVANNAH Agreements have proved very useful in practice. After the initial stage of operation for demonstration and goodwill purposes, the SAVANNAH was transferred from operation by a Government Agent to a private firm (FAST, First Atomic Ship Transport, Inc.) and the indemnification was transferred from Maritime Administration to FAST. This required adaptation of the bilateral agreements concluded with some governments, which was easily done by exchange of letters. Two North-African countries did not even insist on formal bilateral agreements, but accepted the SAVANNAH into their ports on the basis of diplomatic notes [8].

In a number of countries, the conclusion of the SAVANNAH Agreements with the United States of America made amendments to already existing Atomic Energy Acts necessary or required promulgation of new legal provisions concerning nuclear ships [9]. In some cases this created unforeseen legal problems. For instance, in Japan, where a visit of the SAVANNAH had been expected as early as 1965, it was discovered in early 1967 that national liability law applied only to Japanese nuclear ships but not to the SAVANNAH. Despite the fact that there were no safety problems involved at all, the ship could not visit Japan because there was no law that would have been applicable had nuclear damage been caused by the ship [10].

The negotiation and conclusion of SAVANNAH Agreements with some countries may have been difficult and may have taken a considerable time. But at any rate these negotiations certainly stimulated thinking on nuclear law in many countries. This is easily seen when one compares the early drafts with the Agreements in their final form.

4. OTTO HAHN AGREEMENTS

When the Government of the Federal Republic of Germany began to prepare for international voyages of the N.S. OTTO HAHN, a bulk carrier, it was first intended to use the ship for carrying ore from Norway to ports in the Netherlands and Belgium. The first proposal for an agreement was therefore submitted to the Government of the Netherlands; this followed closely the pattern set by the SAVANNAH Agreements. That Government, however, proposed to regulate the problems of third-party liability by including in the agreement the relevant provisions of the Brussels Convention on the Liability of Operators of Nuclear Ships. This was accepted by the German Government and the formal agreement was signed on 28 October 1968. It was ratified on 4 June 1969 by the German Bundestag [11]. Due to unforeseeable delays in parliamentary procedure, ratification by the Parliament of the Netherlands was terminated only in December 1970. After exchange of the instruments of ratification, the Agreement came into force on 18 March 1971. Agreements similar to that concluded with the Netherlands were negotiated with some other countries. Two of them have been signed and are in the process of ratification.

The fact that the relevant liability provisions of the Brussels Convention of 1962 were included by reference into several such bilateral agreements amounted almost to a piecemeal ratification of that Convention. That proved to be a disturbing thought to certain legal minds in Germany because the Brussels Convention, like other international conventions in the field of civil liability for nuclear damage, embodies the principle of 'legal channelling' of liability onto the operator of the installation. The German Atomic Energy Act, however, follows the principle of 'economic channelling' of liability. For this reason the German delegation had made a reservation to the Paris Convention on Third-Party Liability in the Field of Nuclear Energy. It did not seem appropriate now to introduce legal channelling into the German legal system 'through the back door'. A new model bilateral agreement for visits of the OTTO HAHN to foreign territorial waters and ports was therefore elaborated which, in regard to liability for nuclear damage, referred to the national law of the host state. This might have looked attractive to potential host governments. It proved difficult, however, in certain cases, because it was not always easy to establish what exactly the national law of some far-away countries might be, which might also possess different legal systems. In one vital aspect there was similarity with the earlier draft model agreement, namely in that the operator's liability was limited to DM 400 million - an amount roughly equivalent to the 1500 million gold francs of the Brussels Convention. Negotiations based on the first or the second type of draft model agreement were initiated with more than a dozen countries.

Very early in 1970 the Government of the Federal Republic of Germany succeeded through diplomatic channels in arriving at an agreement with the Government of Morocco for visits of the OTTO HAHN to Casablanca and

Safi, where she took on her first cargo of phosphate. Similarly, by exchange of diplomatic notes, an agreement was reached with the Government of Iran. The ship visited the port of Bandar Abbas in early May 1970. Additional agreements were reached with the Governments of Ghana, Mauretania, Senegal, Sierra Leone, Togo, and Tunisia. Invitations for visits of the OTTO HAHN were also received, through diplomatic channels, from Columbia, Peru, and Venezuela.

With Argentina an agreement was negotiated in March 1971. It differs from earlier agreements not so much in its contents as in its form. The provisions concerning the safety aspects and the operation of the ship coincide to a large extent with the regulations of the SOLAS Convention. But in regard to liability for nuclear damage, the agreement does not, as did the German-Netherlands agreement, include the liability provisions of the Brussels Convention by reference. It rather incorporates them more or less verbally to the extent necessary for a bilateral agreement. This method proposed by the Argentine delegation resulted in a rather clearer and much shorter text in comparison with earlier agreements. Since ratification will take some time in Germany, it was agreed on signing the agreement to exchange diplomatic notes that would permit a courtesy visit of the OTTO HAHN to Argentina even before the formal agreement entered into force.

In the light of the experience and the problems encountered in three years of negotiating bilateral agreements, there is consensus that a world-wide legal basis for settling the delicate problems of liability for nuclear damage caused by a nuclear ship can be found only in the 'legal channelling' of liability. Therefore the reluctance regarding this system is waning. The competent government departments are now preparing a draft ratification law. It is hoped that it will be approved by the Council of Ministers in the course of this year (1971) and that ratification proceedings can then begin in parliament.

Ratification by the Federal Republic of Germany of the Brussels Convention would mean that it entered into force. Portugal has already ratified, and Madagascar and the Democratic Republic of the Congo have acceded to the Convention. With the German ratification it would then come into force for the four countries mentioned. Entry into force would certainly induce other countries also to ratify. Some of them, mainly in Europe, are known to be already preparing ratification laws. Another interested country is Japan, which has a nuclear ship, MUTSU, under construction. For Japan to become a party to the Brussels Convention would have several advantages. Under present Japanese law there is no limit to the liability of the operator of a nuclear plant. Under the Brussels Convention, however, this liability would be limited for the operator of the nuclear ship to an amount of roughly \$100 million. This would apply both to the MUTSU and to foreign ships visiting Japan. The MUTSU would also be covered by Government guarantee when visiting foreign ports.

5. EVALUATION, CONCLUSIONS

Let us now try to evaluate the experience gained in the operation of the SAVANNAH and the OTTO HAHN in relation to international legal

provisions applying to nuclear ships [8, 12].¹ This evaluation has to be made from two aspects, the first of which is that of the nuclear safety of the ship. In this respect it appears that most major problems have been settled by the provisions of Chapter VIII of the SOLAS Convention and the Recommendations contained in Annex C to the Final Act of the SOLAS Conference. Both ships were constructed and are being operated in conformity with SOLAS provisions. But, while the SOLAS Convention generally regulates the question of 'seaworthiness' of ships, it may be debatable whether the provisions contained in Chapter VIII have also fully solved the problems connected with the 'portworthiness' of nuclear ships. This problem was, according to my personal notes, already recognized by the French delegation in the Committee on Nuclear Ships during the London Conference in 1960. This means that while there may be no doubt about the fitness of a nuclear ship for an international voyage, there may be problems of a special kind in accepting the nuclear ship into a port.² It was hoped at the Conference that these problems were solved by requiring the nuclear ship to carry a Nuclear Ship Safety Certificate (Reg. 11) and to submit the Safety Assessment. In practice it has turned out that some countries hesitate to accept a nuclear ship even though it carries the Certificate. Before the visits of the SAVANNAH to European ports, the European Atomic Energy Community (EURATOM) established a committee of experts to evaluate the Safety Assessment of the ship. This was done because some member countries felt that they had not sufficient expertise to do this evaluation on their own. During the Brussels Diplomatic Conference on Maritime Law in 1961 and 1962 there were also proposals to include in the Liability Convention provisions regulating the admittance of nuclear ships [13]. These requests were not successful, however. Proposals for an international Licensing System had already been made earlier in the literature [14]. They were not taken up for consideration in practice, probably because they looked good on paper, but turned out to be too difficult to be followed in practice.

The second aspect for evaluation is that of liability for nuclear damage. The bilateral agreements which have so far governed this field, as the Brussels Convention on Liability of Operators of Nuclear Ships is not yet in force, have used two ways. One is the express or tacit reference to national law of the host state. This, of course, results in a good many different liability laws becoming applicable to one and the same nuclear ship, depending on what countries she visits. Making the national law of the host state applicable may not create major problems if this host state has enacted a special atomic energy law; it would normally and clearly regulate problems of liability for nuclear damage. The legal situation becomes more doubtful if the national law does not have such special provisions for nuclear liability, which means that the reference in the bilateral agreement is to common tort law. In many countries this would require a victim of nuclear damage to prove the operator of the ship at fault (negligent), a task which may frequently be impossible to accomplish. The most difficult problems would arise in countries with a federal constitution, if the federal government concludes a bilateral agreement on visits of a nuclear ship and where the liability law is state law.

¹ Information on operating experience with 'Lenin' is unfortunately not available.

² This decision lies with national authorities and national - not international - law.

A number of bilateral agreements refer to the Brussels Convention on Liability of Operators of Nuclear Ships. This may occasionally create minor difficulties because the Convention is written as a multilateral instrument, whereas there are only two partners to the agreements in question. Such difficulties were avoided in some cases by including the relevant provisions of the Brussels Convention almost verbally to the extent necessary for a bilateral agreement. But even this method will always raise questions as to whether the texts taken over from the Convention are complete or whether there are any lacunae left.

At any rate it should be kept in mind that we are discussing legal provisions applicable to seagoing ships. Such ships are built for operation on the seven seas, that is to say on all sea-lanes of the world. It would therefore be illogical to subject them to a series of bilateral legal arrangements which are partly done as formal state treaties, partly on the basis of mere diplomatic notes. There seems to be only one sensible approach to achieving uniform treatment in all countries and for all nuclear ships. This is to apply to them, in respect of liability for nuclear damage, a world-wide regime. Such a regime can be at the disposal of all seafaring nations and of all nations that have ports to accept ships on international voyages, in the form of the 1962 Brussels Convention on Liability of Operators of Nuclear Ships.

There appears now to be a good chance that this Convention will come into force soon, with ratification being actively considered by the Federal Republic of Germany and Japan. It is hoped that an increasing number of countries would then also decide to ratify, so that the Convention would gradually begin to weave a network of an applicable and practicable international law across the oceans.

Entry into force might also lead to considering an amendment to the present text of the Convention that would result in the exclusion of warships from its scope of application. The United States of America and the USSR would then no longer have an excuse for staying outside the Convention. It should not be too difficult for legal experts in this field to propose practicable and easy ways for such an amendment. Once the Brussels Convention becomes acceptable to all states that have licensed civilian nuclear ships, there should no longer be a need for bilateral agreements relating to visits of such ships to foreign ports.

The aspects of nuclear safety are already fully covered by the SOLAS Convention. The liability problems will be resolved through the Brussels Convention, to which, it is hoped, a great number of countries will become parties. What will be left are minor problems, such as local arrangements for different ports and, perhaps, different ships. But they can be solved without difficulty between local authorities and operators of nuclear ships without any need for further law-making.

The international organizations can also play an important part in making nuclear ships more easily acceptable in foreign ports. The International Atomic Energy Agency (IAEA) has already done something in this respect, by publishing a report prepared by a panel of experts on 'Safety Considerations in the Use of Ports and Approaches by Nuclear Merchant ships' [15]. The IAEA is also, I understand, willing to assist Member States, upon request, in the evaluation of safety assessments of nuclear ships.³ On the other hand, IMCO could, in my opinion, prevail

³ As EURATOM has done (see paragraph 1 of section 5).

on governments of Member States to accept Nuclear Ship Safety Certificates issued by the government of an other Member State at face value, thus attaching to them the function given to these Certificates by the SOLAS Convention. After all, the Certificates are issued to the operator of the nuclear ship by a responsible authority and are an official statement that the ship complies with the regulations for nuclear ships laid down in the SOLAS Convention. In other words, this means that the nuclear ship can be considered as safe. It may be well understandable that certain people and certain governments feel still reluctant vis-à-vis the 'unknown' in nuclear ships. But since the SAVANNAH has been in operation for several years and the OTTO HAHN has been operated successfully for more than two years, the 'unknown' should gradually begin to be considered as well proven and acceptable. This is already the case in those ports at which the SAVANNAH or the OTTO HAHN have called regularly or several times in the course of their commercial operations. There is hope that, as time progresses, this will also be the case in many other ports and countries. Just as there is enough technical expertise to make the safe construction and operation of nuclear ships a reality, there are sufficient legal provisions to regulate the operation of nuclear ships on the high seas and in foreign territorial waters and ports [16-21].

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DISCUSSION ON AGENDA ITEM 3.4

Legislative, insurance and regulatory aspects

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

- P/767 OECD Presented by P. Strohl*
P/624 France Presented by B. de Nercy
P/089 USA Presented by H.K. Shapar
P/043 USA Presented by M. Joslin

L. FAILLA: As Mr. Strohl doubtless knows, a problem currently under consideration is the exclusion from the provisions of the Paris Convention of small amounts of nuclear material — whether in transport or in use within an installation — which are considered to present no serious danger. I should be interested to know what criterion is being applied (perhaps in terms of dose) to define the concept of 'serious danger'. I am familiar with some documents on the subject, but they are all concerned with accidental releases from a reactor, not with ordinary utilization or transport of nuclear materials. I gather, too, that the amounts being considered for exclusion are not the same under the Paris and Vienna Conventions.

P. STROHL: This is, of course, a rather technical question. The proposals for exemption are based, in the case of both Conventions, on the recommendations of experts. But the limits adopted will necessarily be somewhat arbitrary, for it is extremely difficult to judge exactly where the limit should lie — what risk is just great enough to justify application of the exceptional regime laid down in the Conventions rather than the normal provisions of public law. Lawyers and scientists will have to hold careful consultations on this matter in order to arrive at a reasonable solution.

The exemptions for transport at present in force are the same under the two Conventions. There are, as you say, certain discrepancies between the new proposals of ENEA on the one hand and IAEA on the other for exemptions applicable to materials within nuclear installations. However, the two Agencies will undoubtedly do their best to reach final decisions which are identical or, at the very least, compatible.

E. MAURER: I am familiar, of course, with the protection which the Paris and Vienna Conventions afford to the victims of nuclear incidents by centralizing jurisdiction and by accepting the doctrine of absolute liability. What are the amounts that would be available for the compensation of victims under these two Conventions?

P. STROHL: The minimum amounts of financial security adopted in the Paris and Vienna Conventions reflect, essentially, the resources of the insurance market. Under the Paris Convention the security should, in principle, be US \$15 million, and cannot in any case be less than US \$5 million; under the Vienna Convention the minimum security is also US \$5 million. The

Paris Convention also envisages the possibility of Contracting Parties taking steps to increase the amount available for compensation, a matter dealt with in the Brussels Supplementary Convention.

P. COSTE: Do the insurance limits correspond to incidents of specific types that have been identified and analysed? Is there any correlation between the calculations of the insurers and safety assessments carried out, say, by probability analysis? In other words, is the establishment of the insurance premium somehow related to the probability associated with a given risk?

P. STROHL: Perhaps Mr. Stein would like to answer that question.

R. M. STEIN: The limits of liability specified in the Paris and Vienna Conventions were, of course, greatly influenced by the extensive technical studies conducted in the United States of America and in Europe with a view to determining the theoretical scope of nuclear incidents of various types. At the same time, account had to be taken of the amounts of money that could be made available on the private insurance market or through other financial security, as well as through State intervention. As far as the Vienna Convention is concerned, the matter of providing substantial amounts of security — because it is a world-wide Convention — was of particular concern to developing countries. Thus, the amounts established were inevitably a compromise between technical estimates of potential damage and the extent of the funds which could be made available.

With respect to insurance premiums, they are based in each instance on a detailed evaluation of all aspects of the risk to be covered.

F. C. BOYD: I should like to ask Mr. de Nercy what insurance cover is provided for nuclear power plants in France, and by whom. Is it the Commissariat à l'énergie atomique or some other governmental regulatory agency that sets the limits?

B. de NERCY: The liability of operators of nuclear power plants was established by the Act of 30 October 1968 Concerning Civil Liability for Nuclear Damage. The limit is F. Fr. 50 million, and only 25 million for public organizations. Beyond these limits, coverage is guaranteed by the State up to a limit of F. Fr. 600 million.

E. MAURER: I should be interested to know whether France is witnessing a clamour about the state of the environment similar to that which has arisen in the United States, and also whether the Commissariat à l'énergie atomique has been given authority to consider all environmental factors — not only those related to direct radiation effects.

B. de NERCY: Problems of the environment have in fact begun to attract widespread public attention in France. This new-found concern is reflected in the establishment of a State Ministry responsible for environmental problems and a High Commission for the Environment. However, the procedure for obtaining authorization for nuclear installations under the decree of 1963, in which the Commissariat à l'énergie atomique plays an important part, envisages at present only the nuclear aspects of the matter and the risks associated with them. At the same time I should point out that there are other procedures and technical requirements — affecting building permits, for example, or the authorization to tap water supplies, dump waste water in rivers, and so on — which also regulate applications for new projects.

S. PANOPOULOS: Is there any provision in French legislation for an immediate release of money to the victims of a nuclear incident, before

the insurance companies or the Government are able to make formal compensation?

B. de NERCY: No, there is no special provision which would allow immediate indemnification of victims; but, clearly, the requisite sums would be released very quickly in urgent cases.

L. FAILLA: I have a question for Mr. Shapar. What federal authorities in the United States issue authorization for the use of equipment containing artificial radioisotopes produced either in reactors or in accelerators?

H. K. SHAPAR: The United States Atomic Energy Commission (USAEC) licenses and regulates not only facilities such as nuclear reactors but also source materials (generally uranium and thorium), by-product materials (radioisotopes produced in reactors) and special nuclear materials (reactor fuel). The individual States may license and regulate such materials provided they enter into an agreement with the AEC for that purpose. The AEC does not license or regulate radium or cyclotron-produced isotopes. Apart from the licensing of the production and utilization facilities and materials I have referred to, there is no federal licensing of other equipment from the standpoint of radiological protection.

DISCUSSION ON THE FOLLOWING GROUP OF PAPERS:

P/400 FRG Presented by N. Pelzer

P/401 FRG Presented by W. Boulanger

There was no discussion on these papers.