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NUCLEAR SAFETY AND RADIOLOGICAL PROTECTION

CONTENTS

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CONTENTS

THE FEDERAL REPUBLIC OF GERMANY'S APPROACH TO NUCLEAR SAFETY AND RECENT OPERATING EXPERIENCE

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Contents

Development and Present Status of NPP's in the FRG

. Development

The year 1955 marked a historical cornerstone: In this year the first United Nations International Conference in Geneva on the Peaceful Use of Nuclear Energy took place and the FRG signed the Paris Treaty, taking on the obligation to use nuclear energy only for peaceful purposes. It is the year where first preparations in the FRG started for the utilization of nuclear energy. Five years afterwards the Atomic Energy Act of the FRG became effective (January 1960) and the first commercial nuclear power plant was commissioned at Kahl in 1960. Thus, electricity has been produced in the FRG by nuclear energy for more than 25 years.

The development of commercial light-water cooled nuclear power plants began with the construction under licenses of U.S. vendors. Very soon independent German developments were established. The result was the rapid introduction of light-water reactors, both BWRs and PWRs, of German design.

In this context, it is worth mentioning that only large utilities in the FRG operate or bear responsibility for operation of NPP's thus providing the necessary resources, expertise and infrastructure. Also, essentially all nuclear power plants in the FRG are turnkey projects. The state of design and planning is usually highly advanced before construction begins. It is furthermore important to note that licenses are usually granted both to the utility and the vendor. The vendor operates during commissioning the plant on its own responsibility from first criticality via various power steps up to full power before commercial operation and take-over by the utility. There is - with one exception - only one vendor. Both vendors have shown their capacity in efficient construction and operation of NPP's.

Nuclear power plants have seen substantial development and considerable maturing process. The size of the plants was increased and the safety concept was adapted to the advancing state of science and technology.

Worth mentioning are the following steps: the 300 MWe-, and (600-900) MWe PWR and BWR classes in 1966-1968, and 1972-1979, respectively, the 1300 MWe Biblis PWR class in 1974-1978, the 1300 MWe BWR class in 1984-1985, and the 1300 MWe Preconvoi-PWR class in 1981-1984.

. Status

Considerable efforts for standardization, close cooperation and acceleration of licensing have been taken in the FRG. To this end, the Konvoi-project has been established in 1982. This project is realized in three Konvoi plants which are currently under construction. The construction period has been reduced by about 20 % relative to the Prekonvoi power plants. Commissioning and erection on site is well on or ahead of schedule. At two of the three plants structural work on the buildings is finished and the pressure and leak rate tests for the primary containment vessel were successfully conducted. The pressure testing of their primary systems will be completed this spring. The time difference of the third Konvoi plant with respect to both other Konvoi plants will be half or one year. Two of the Konvoi plants will start commercial operation presumably in 1988, the third plant in 1989.

Altogether, 21 units are presently in operation with an installed total net electrical power capacity of about 19 000 MW. 4 units are under construction with an additional net capacity of about 4000 MWe. Including plants under construction the share of BWRs and PWR is about 30 % and 67 %, respectively, showing that other reactor types like gas cooled reactors or the fast breeder constitute only a minor fraction of the total nuclear capacity. The nuclear energy's share of electricity generation in the FRG will increase from slighly more than 30 % today up to about 40 % in 1990. Whether or not new nuclear power plant projects will be realized in the future cannot be foreseen at the moment.

Five demonstration plants already in operation were shut down partially after 20 years or even 25 years of successful operation. They are in various steps of decommissioning. For one plant license for total dismantling has been granted recently.

Fig. 1 shows sites of nuclear fuel cycle facilities in the FRG. A compilation of NPP's in the FRG can be found in Table 1.

. Performance

Nuclear power plant performance is very satisfactory from an economical point of view. FRG units have been ranked constantly amongst the worldwide top level power plants in recent years.

3 FRG nuclear units produced each more than 10 Mio kWh in 1984 and 1985, and 2 units each in 1986, respectively. The good performance is further demonstrated by the average capacity factors of 82,8 %, 83,4 % and 80,4 % in the last 3 years, 1984, 1985 and 1986. The FRG had 12 of its 19 units running above 80 % capacity in 1986, and in the preceding years 14 out of 18 (1985) and 9 out of 17 (1984). Only one unit, a demonstration plant performing specific tests, was below 50 % capacity in the last 3 years. The high availabilities of the BWRs have been reached only recently after extensive backfitting measures have been terminated (exchange of most of the primary pipes).

Non-availiabilities of capacity due to unplanned outage times are on the average less than 3 % in the last 3 years, an appreciable fraction of the units are below 1 %. More than 70 % of these forced outages in LWR plants have been caused by steam generator problems and conventional equipment such as turbine generators, main heat removal systems, feedwater and condenser systems. The outages caused by nuclear components failures have been less significant.

The trend towards improved operating performance is further demonstrated by the relative small number of reactor scrams . The plants have reached a level of about 1.5 scrams (on the average) per plant and per year in the last five years.

Operating Experience

. Reporting System

The availability of information on the national level is mainly assured by a formalized reporting system. Already about 10 years ago the competent authorities agreed to install a centralized reporting office in the FRG which was subsequently established within the Gesellschaft für Reaktorsicherheit (GRS), mbH. Licensees are required to report abnormal occurrences according to a set of well defined reporting criteria. According to their safety relevance and required urgency of administrative action the occurrences are grouped into different categories:

Category S (S=sofort=immediate) Abnormal occurrences requiring immediate steps of the competent authority. These occurrences include those events which show acute safety relevant deficiencies.

Category E (E=eilig=urgent) Abnormal occurrences for which the causes must be clarified and abolished in due time. These are e.g. occurrences of potential safety significance without, however, requiring immediate actions.

Category N (N=normal) These are events of general interest about which the authority must be informed. As a rule, these are events without being immediatly or potentially significant from the safety point of view which, however, go beyond normal operation.

Category V (V=Vorsorge=precaution) These are events which occurred during the construction period of the plant before commissioning and about which the competent authority must be informed in view of the subsequent operating phase of the plant.

The classification of an event is done according to an (preliminary) assessment at its occurrence.

Depending on the category the abnormal event must be reported from immediately to two weeks after its occurrence. The storage of all events is done by a data bank. Currently the data bank contains about 2500 documents.

Experience from abroad is made available by participation in international cooperations. The FRG participates in the IAEA-IRS through the NEA-IRS. Besides, bilateral arrangements exist with a number of countries.

. Exposures

From a safety point of view operating experience with nuclear power plants in the FRG is positive. No accidents resulting in a hazard for plant safety, personnel or environment have occurred within the last ten years. Also, none of the reported abnormal occurrences could be considered alarming from the point of view of safety in the way that a loss of essential safety features had been experienced. Within the last six years only one event had to be classified as potentially radiologically significant belonging to the highest reporting category. Here, the admitted daily release had been exceeded. However, the totally released annual activity was far below licensed values. Control measurements in the surroundings of the plant showed no increased radiation exposure.

Moreover, it is noteworthy that maximal doses to the public even from abnormal events which have occurred so far in the FRG have never exceeded limits set for normal operational releases (see appendix), thus being smaller than the variation of the natural background radiation.

Operational releases of radionuclides via gaseous and liquid effluents have continuously been reduced for the individual plants. Although the nominal electrical power output and the number of operating nuclear power plants has been increased considerably, the accumulated releases are fairly constant over the years showing thus a significant reduction per produced electrical energy. Doses to the public due to routine emissions have been calculated using pessimistic assumptions. The doses are negligible and far below admissible values. Typical calculated committed annual doses in most unfavorable cases are 1,0 μSv and less. These doses constitute an insignificant fraction (less than 0,1 %) of the annual natural background radiation.

. Statistics of abnormal occurrences and trends

On the average, 10 - 20 abnormal occurrences requiring reporting were recorded in recent years per year and per plant. Typically, most of the events fall in the lowest reporting category ($\sim$ 80 %). About one fifth of the reported events fall into the middle reporting category. No event falling into the highest class has been reported in the last 3 years, 1 event in the last 6 years.

The majority of the events did not cause a restriction of the power production (70 % and more), a vast part of them was discovered during routine inspection, during maintenance or repair works. About 30 % are typically caused by component failures. The same number holds for faulty repair, maintenance, operating modes or operating conditions. Design or manufacture faults constitute a slightly smaller fraction ($\sim$ 20 % alltogether).

The overwhelming number of failures is corrected by repair works and only to a lesser extent by modifications, showing that in most cases the cause is just normal wear and not constructive deficiency.

. Evaluation and feed-back

The effectiveness of the feedback obviously depends on the technical details released by the utilities and on the effort spent on the evaluation of the experience. Our experience is that sufficient manpower of high qualification has to be devoted continuously to this task. In the FRG the evaluation is done on the federal level by GRS on behalf of the Ministry of Environment, Natural Protection and Reactor Safety. It is important to point out that the licensees also need adequate analytical capacity to make use of information on operating experience from other plants.

A thorough evaluation is also a prerequisite for the preparation of a meaningful input for the international cooperation. In the FRG, GRS evaluates German as well as foreign experience and prepares reports for international exchange.

The distribution of information on national and international level is done by periodic and ad hoc reports. All authorities concerned with nuclear safety, their advisory bodies, the utilities, the plant vendors, expert organizations and component manufacturers (if concerned) are included in the information exchange. For international exchange, reports are submitted to international Incident Reporting Systems. Moreover a direct information exchange takes place with a number of countries.

The most essential part of the feedback of course is, that appropriate actions are taken as a consequence of experience. It is our view that permanent attention to operating experience and an ongoing technical discussion are more important than formal regulations. In the FRG, a large number of improvements were introduced as a consequence of operational experience.

. Backfitting

In the Federal Republic of Germany, backfitting practice may be conceived as a pragmatic improvement of the actual state of the plant. Decisions are usually taken case-by-case paying due respect to all relevant aspects. By consideration of operating experience and insights of R+D-results, the safety of the plants is continuously improved. Here, also experience in foreign plants resulted in actions in German plants.

The overwhelming number of these backfitting measures was implemented voluntarily by the utilities after discussions in the sientific-technical area. One has always succeded this far to convince the utilities of the necessity of such measures. Also, very often the utilities have proposed backfitting measures as a result of their own investigations. The utilities filed the applications which had to be licensed.

An initiative of the utilities in the implementation of new safety related findings is to be welcomed. It leads to faster results than would be the case with imposed provisions.

A large number of plant alterations have been implemented. Typical examples are:

- Inspections in BWR's revealed deficiencies on the main-steam and feedwater pipes. Also, compared to the construction period, higher-grade material characteristics and optimized manufacturing techniques became available in recent years. A large number of main-steam and feedwater pipes has been exchanged in German BWR's.

- In recent years, spatial redundancy of systems and the independence of redundant systems for the control of external events received greater attention. Most of the older PWR's have been retrofitted with a bunkered emergency system.

- In an older PWR, operating experience showed that radiation embrittlement of the reactor pressure vessel might be higher at the end of its operating period than demanded by present requirements. Operating modes have been changed as countermeasures against radiation embrittlement. Moreover, the high pressure injection system has been modified to preclude brittle fatigue in the event of cold water transients.

- Backleakage of nitrogen from the accumulators through leaking check valves resulted in failure of a safety injection pump to operate during a test. The piping arrangement was changed in several plants.

- Diverse pilot valves for pilot operated safety and relief valves were installed based on experiences with common cause effects in German and foreign plants.

- It has become evident that steam generator tube failures are not very rare events. Contaminated steam would be released to the environment via relief valves in the event of tube failures and loss of the main heat sink. Several measures have been carried out, e.g. increase of set points of the secondary side relief valves and modifications of the operators procedures, to reduce possible releases of radioactive nuclides to the environment.

Safety Concept

. General aspects

Nuclear power plants are designed against disturbances and incidents in order to assure the safe shutdown, the coolability of the reactor core, the residual heat removal and the retention of fission products. The reactor safety concept in the FRG is basically the same as in other countries. It is based on physical barriers to retain the hazardous radioactive materials and on the concept of defense-in-depth to protect the integrity of the barriers against consequences of incidents and accidents.

Compared to other countries there are, however, some peculiarities in the FRG.

Inherent feedback-characteristics of the plant, redundancy and - if possible - diversity and especially physical separation of safety related equipment are vital elements in our safety concept. Also, the high degree of automation and the single-failure and repair-criterion are important issues in this respect.

. Redundancy and automation

Safety-related subsystems are redundant and are not only functionally separated but also physically and are constructionally protected. In our concept, interlacing between redundant subsystems, including their auxiliary devices like automatic control and power supply, is avoided as far as possible. Because of the single-failure concept and the additional postulation of a repair job on another redundancy, there are two more redundancies than are needed to cope with an incident ("n+2"-system layout).

These are valuable design tools to combat common-mode failures. Also, in-service inspections of individual trains can be carried out on a train-by-train basis in greater depth, without interference with necessary automatic safety measures.

Redundant safety control systems are provided which are separated from other systems. These control systems control all safety-related functions. Priority circuits ensure that the signals of the safety control systems take priority over other signals, particularly over the operating controls and manual actions.

Manual measures to cope with incidents are only required after a period of 30 minutes. The necessary safety measures are automatically initiated at least during the first 30 minutes following an incident. The power plant is automatically transferred to a safe follow-up state. The effectivness of manual measures are indicated in the control room. Also, whenever such actions are required, the respective signals occur in the control room and appear on the main control desk, the system control desk, the reactor protection panel and on the computer annunciation units.

As a result of the far-reacting automation concept, the major task of the responsible control room personnel is monitoring the state of the plant. Such a layout helps to avoid human errors, particularly in stressfull situations. It is a step forward towards an error-tolerant reactor.

. Progressive limitation and protection by control systems

Moreover, to reduce the actions of the safety control systems and the workload of the operators, a concept of defense-in-depth has been applied with respect to reactor protection. This concept is characterized by a hierachical structure of automatic operating controls, limitation systems and the protection system.

The first level of defense is automatic control. The major process variables are controlled, with the inclusion of dead bands, so that they remain constant or at given values dictated by the load concerned.

The next line of defense is realized by so-called limitation systems. There are two different types: Condition limitation and protection limitation. The first ensures that process values do not exceed the limits specified in safety analysis. The other is designed for protective countermeasures for specified events. Limitation measures take priority over operating controls and preceed the protective actions of the safety systems.

The protective actions of the safety systems constitute the third level in this progressive concept. These actions are initiated in cases which are not coped with by operating controls or limitation units. Actions at this level override actions at preceding levels.

With this hierarchical concept a more appropriate and flexible response to disturbances and incidents is achieved.

The low number of reactor scrams and spurious initiations of safety systems demonstrate the effectiveness of this concept as a whole.

In spite of this, the shift personnel has operating procedures available to ensure that incidents can be manually controlled. The staff is given training and periodic retraining. Here, as a back-up, also flexible use of safety features in the plant are given due weight to halt accidents in case of unpredictable event sequences or sequences which have not been previously considered in the design. This utilization of safety margins which is hidden in the present design is part of the German safety philosophy.

. Incident and accident prevention

The German safety concept gives priority to measures for incident and accident preventions, supplemented by additional measures for accident mitigation.

Typically German PWR's are thus equipped with a spherical full-pressure (dry) containment of steel which can withstand the pressure build-up after loss-of-coolant accidents. Spray systems to limit the containment pressure in the event of a total mass and energy discharge of the primary loop are not necessary. Connections of the containment to the outside are secured by redundant isolating devices. The containment is surrounded by a concrete shell, with a wall thickness up to about two meters. Leakages from the containment into the annulus between the steel and the concrete shell can be exhausted and filtered for controlled emission through the stack. The concrete shell protects also against hypothetical external events like aircraft crashes and shock waves from chemical explosions. Crash loads from a fastflying jet-fighter including induced vibrations are used as design basis.

Systems necessary to assure decay heat removal, reactor shutdown, and long-term subcriticality are located in a separate building which is also protected against external impacts. Stored coolant and available energy supply are sufficient to keep the plant in a safe state for at least 10 hours without manual actions.

In this bunkered system an emergency control room is integrated which gives full control of the shut down systems and the residual heat removal system in case the main control room is not available due to external events.

Moreover, due to very stringent requirements in the FRG also to protect nuclear power plants against external events, vital system functions like emergency power and emergency feedwater supply are installed with higher redundancy and higher diversity than compared to other countries. So, in addition to the main feedwater system an auxiliary feedwater system which can operate on emergency power has been installed. Furthermore, a complete independent emergency feedwater system does exist with four trains, both physically and functionally separated, each with autonomous water and power supply, and each being dedicated to different steam generators. In this way not only the reliability of the heat removal via secondary-side systems was increased to cope with external events but also in view of small leaks and operational transients.

Another typical example of the prevention principle is the concept of "basic safety" which has been developed to avoid loss of coolant accidents. This concept is based on high-grade material characteristics, particularly by limitation of the contents of metal impurities and trace elements, and by application of optimized manufacturing techniques. Also, the number of welding seams are minimized and welding seams are localized outside areas of increased stress. This concept aims at high toughness and ductility of the pressure retaining boundary supplemented by increased testability and resistance of the piping against unlikely, yet undiscovered, cracks.

The basic safety concept includes not only the primary coolant and connected systems, but also the pressure retaining walls of pipes, fittings, valves, pressurizers and pumps of external systems with importance to safety.

The improvement due to the basic safety concept permitted a modification of the LOCA-philosophy: 10 % of the main coolant pipe cross section (0.1A) could be defined as a representative design break-size with respect to reaction and jet forces on walls, pipes and components.

This approach resulted in the deletion of unnecessary pipe whip restraints, which hindered recurrent inspection. Radiation exposure of personnel has significantly be reduced for this type of work.

On the other hand, especially to maintain the independence of safety barriers, the double-ended break postulate (2A) was kept as design basis for the effectiveness of the ECC-Systems and, furthermore, to demonstrate that the containment and its retention devices sufficiently limit the release of radioactive materials under the pressure and temperature conditions prevailing also in the event of additional fuel rod cladding damage.

<u>Use of probabilistic methods and risk oriented safety goals</u>

. PRA in licensing and decision making

Today, the probabilistic approach to reactor safety is more widely accepted. The methods are used worldwide to assess the technical safety supplemental to the traditional safety analysis.

In the FRG, the probabilistic methodology is within licensing an established and well-proven, however, informal and pragmatic tool. The probabilistic assessments are restricted to specific items concerning design basis events and safety functions to cope with these events, mostly with regard to core coolability and to a lesser extent to containment isolation.

No formal guidelines or quantitative safety targets have yet been established. Quantitative figures evolve as a matter of practice. These figures are used for orientation purposes. An assessement of the quantitative results is not the main objective. Rather, weak point identification and optimization in view of a balanced safety concept are primary issues.

Given the uncertainties of analytical probabilistic methods extremely low frecuencies are always to be looked at with great caution. This is always the case when very remote accident scenarios have to be considered.

Many experts express the view that in such situations a fundamental limitation of the probabilistic approach becomes evident, which - to a large extent - has its roots in a gap of knowledge.

PRA are thus used in a process in which conventional approaches retain their dominant and continuing value.

. Full-scope risk studies

It is an independent aspect, however, that severe accidents beyond the design limit were and are still being investigated in in-depth research programs. In this context complete risk studies are performed in the FRG. Also, plant-specific probabilistic studies which are confined to selected issues have been completed or are underway.

The results are carefully analyzed by industry, authorities and legal bodies. As a matter of fact, changes in the design or operation were and will be stipulated.

In the FRG, accident management measures are investigated and internal emergency response actions as a back-up of design provisions which fully utilize the safety margins in present design, particularly the avoidance of a large scale coremelt under unfavourable conditions and the fission product retention facilities of the containment. Also, specific measures in the event of core melt accidents for controlled filtered depressurization of the containment for LWR are considered.

The results of the Phase B of the German Risk Study will help us in decision making.

Risk studies are thus used in an operational way to improve and optimize the safety of nuclear power plants.

. Safety goals

By risk oriented safety goals a set of quantitative criteria is meant against which the safety level of a plant can be judged, using probabilistic methodologies. It is a subject of controversial discussions which would be the appropriate form and how these criteria should be set.

Different approaches have been considered so far. The proposals are still being debated. We are in the FRG not as close as other countries to formulate a safety goal policy.

Outlook and future prospects

The third decade of the peaceful use of nuclear energy in the Federal Republic of Germany has seen a phase of consolidation and a considerable maturing process. Operating experience in the FRG is positive. The methods of reactor safety have proven to be satisfactory. Therefore, the basic safety principles in the FRG can be expected to remain in its present form for an extended period.

Nevertheless, safety engineering will not stand still. In the FRG, there is a basic requirement to continuously adapt nuclear safety to the advancing state of science and technology.

At present there are concrete plants to expand the safety concept to include severe accidents. Measures are discussed aiming at intervention during the course of an accident even beyond the design basis.

In the FRG, those measures should be taken in a pragmatic way if, with acceptable effort, a substantial reduction of the already small residual risk may be accomplished.

In this context, the following cases can be distinguished

- to assure that design limits of the core are not exceeded although the state of the plant has already passed the design limit as a whole

- to assure complete and long-term coolability also in situations with core heat-up and severe core damage

- to assure emergency containment isolation with increased internal pressure

- to further diminish the threat of containment failure due to hydrogen combustion in LWR's

- to further mitigate the consequences of hypothetical nuclear core melt accidents by controlled filtered depressurization of the containment of PWR's.

In August 1986 the Reactor Safety Commission has requested plant vendors and operators of all German NPP's for an information on plant specific items. Preliminary results of the deliberations have been set forward end of 1986. The complex should be terminated essentially about end of 1987.

The basic philosophy is to flexibly use existing safety and operational systems with due consideration of new findings in nuclear safety research.

In the decison making, the information presented in the course of the evaluation of the Tschernobyl accident and the investigations of core-melt phenomenology in the context of the German Risk Study Phase B, as well as preliminary risk study results concerning other plants, played a substantial role.

First decisions have been taken. One nuclear power plant is already equipped with facilities of controlled containment depressurization.

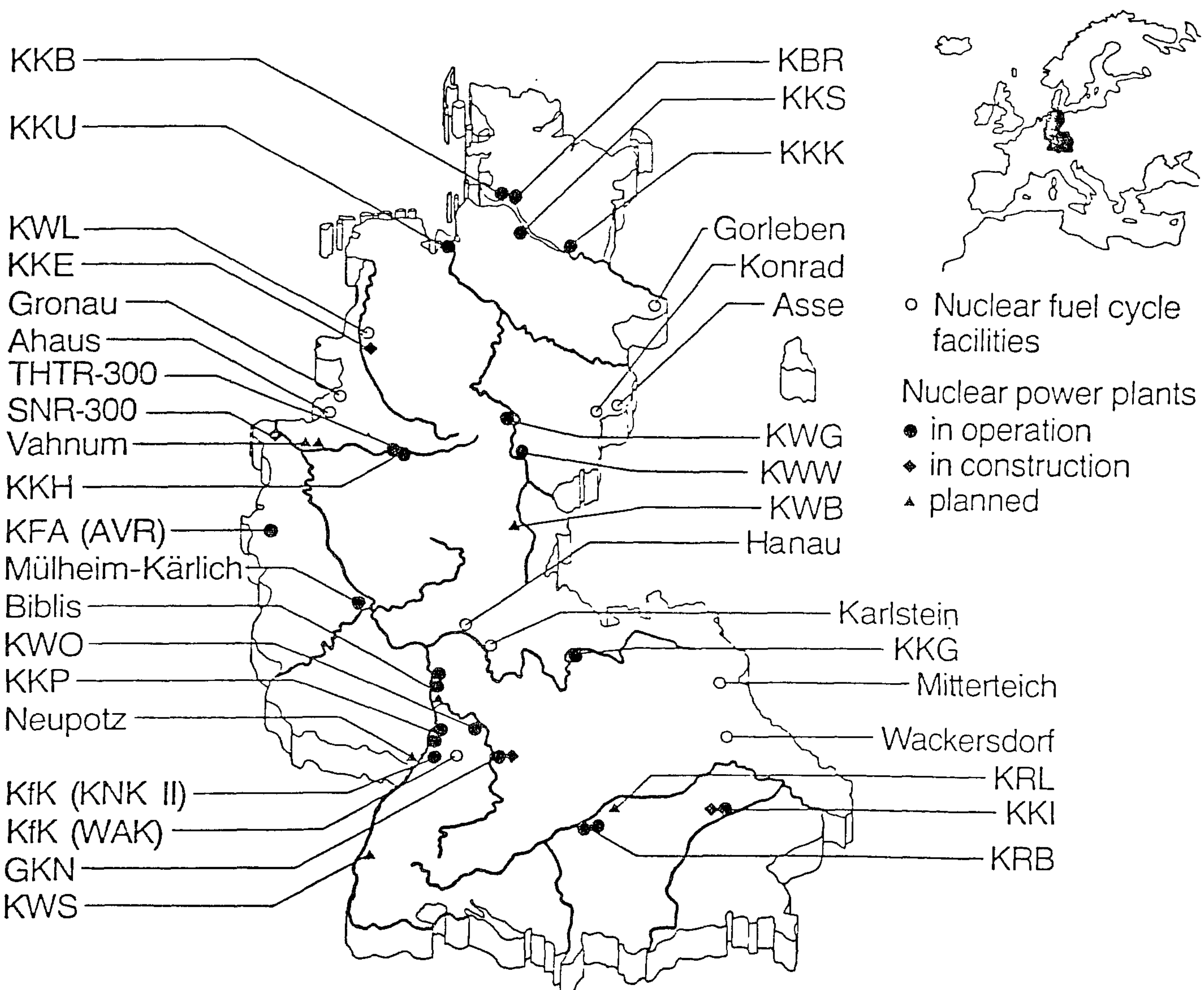

Fig. 1:

SITES OF NUCLEAR POWER PLANTS AND NUCLEAR FUEL CYCLE FACILITIES IN THE F.R.G.

(December 1986)

Table I:

Nuclear Power Plants in the F.R.G. (Dec. 1986)

1. PWR's		Net electrical power (MW)	Start of commercial operation
Obrigheim	(KWO)	340	1968
Stade	(KKS)	640	1972
Biblis A		1146	1974
Biblis B		1240	1976
Neckarwestheim 1	(GKN 1)	795	1976
Unterweser	(KKU)	1230	1978
Grafenrheinfeld	(KKG)	1235	1981
Grohnde	(KWG)	1300	1984
Philippsburg 2	(KKP 2)	1268	1984
Mülheim-Kärlich	(KMK)	1227	1986
Brokdorf	(KBR)	1307	1986
Isar 2	(KKI 2)	1285	1988 *)
Emsland	(KKE)	1242	1988 *)
Neckarwestheim 2	(GKN 2)	1230	1989 *)

2. BWR's		Net electrical power (MW)	Start of commercial operation
Würgassen	(KWW)	640	1972
Brunsbüttel	(KKB)	771	1976
Isar 1	(KKI 1)	870	1977
Philippsburg 1	(KKP 1)	864	1979
Krümmel	(KKK)	1260	1983
Gundremmingen B	(KRB B)	1244	1984
Gundremmingen C	(KRB C)	1244	1985

3. HTGR's		Net electrical power (MW)	Start of commercial operation
Jülich	(AVR)	13	1966
Uentrop	(THTR-300)	296	1986

4. LMFBR's		Net electrical power (MW)	Start of commercial operation
Karlsruhe	(KNK II)	17	1978
Kalkar	(SNR-300)	295	1986 *)

*) under construction

Section 45 - Dose limits for areas which are not radiological protection areas

The radiological protection officer under Sec. 29, para. (1) shall plan the technical design and operation of his installations or facilities in such a way that the radiation exposure of humans resulting from the discharge of radioactive materials in air or water from these installations or facilities is kept as low as practicable and in each case does not exceed 3/500 or, in the case of the thyroid via food chains, 3/1000 of the values specified in Appendix X, Column 2. This radiation exposure shall be calculated for the most unfavourable points of impact and considering all relevant exposure pathways including food chains; the individual assumptions and procedures to be applied in the determination of the radiation exposure shall be established by the Federal Minister in charge of reactor safety and radiological protection by way of ordinance with the consent of the Federal Council. In so far as other installations or facilities at these or other sites contribute to the radiation exposure at the designated points, the responsible authority shall ensure that the total of the values specified in the first sentence is not exceeded.

Appendix X

Body dose limits for occupationally exposed persons

Body part	Category A x) occupationally exposed person per calendar year	Category B x) occupationally exposed person per calendar year
(1)	(2)	(3)
1. Whole body, bone marrow, gonads, uterus	50 mJ/kg (5 rem)	15 mJ/kg (1,5 rem)
2. Hands, forearms, feet, lower legs and ankles including the relevant skin	600 mJ/kg (60 rem)	200 mJ/kg (20 rem)
3. Skin if this is the only part exposed to radiation, excluding the skin on hands, forearms, feet, lower legs and ankles	300 mJ/kg (30 rem)	100 mJ/kg (10 rem)
4. Bones, thyroid	300 mJ/kg (30 rem)	100 mJ/kg (10 rem)
5. Other organs	150 mJ/kg (15 rem)	50 mJ/kg (5 rem)

x) For persons under 18 years, see Sec. 49, para. (2)

The limits referred to in subparas 1 to 5 for parts and organs of the body include the whole body dose limit referred to in subpara. 1.

FILTERED VENTING OF SWEDISH REACTOR CONTAINMENTS

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CONTENTS

FILTERED VENTING OF SWEDISH REACTOR CONTAINMENTS

1
INTRODUCTION AND SUMMARY

Sweden has 12 operating light water reactor nuclear power units - 9 ASEA-ATOM BWR's and 3 Westinghouse PWR's. The BWR's have pressure suppression containments and the PWR's have large, dry containments. Sweden is in a unique position of heavily depending on nuclear power in the near-term (approximately 50 % of total generation) while having decided by public referendum (in 1980) and government decision to abandon it in the long-term. The Swedish government has decided that nuclear power is to be phased out in Sweden by the year 2010.

After the accident at Three Mile Island in 1979 the Swedish Government appointed a Reactor Safety Commission to re-evaluate the risks of nuclear power and to investigate what actions should be taken to enhance safety at Swedish nuclear power plants. The Commission in its recommendations emphasized the importance of continued work to prevent accidents. However, the Commission also suggested that further investigations be made to mitigate the consequence of core-melt accidents involving loss of containment function. In this context the reduction of land contamination was mentioned specifically. Atmospheric venting of the LWR containments via filter systems appeared to the Commission to offer the potential of reducing appreciably the releases considered, and a feasibility study on vent-filtered containment concepts was recommended (1).

Early in 1980 the Swedish Institute of Radiation Protection presented a report titled "More Effective Emergency Planning" in which the consequences of core-melt accidents were studied. Recommendations on improved emergency planning were made, relating to different levels of ambition. The report focused attention on the long-term consequences of a major accident in a nuclear power plant. It was sceptical towards the use of probabilistic estimates of accident risks as a basis for policy decisions; thus greater emphasis on mitigating the consequences was implied (2).

The Swedish nuclear industry was responsive to the recommendations of the Reactor Safety Commission. Work on vent-filtered containment concepts for possible implementation in operating plants commenced in February 1980 in the form of the FILTRA project. It was sponsored by the Nuclear Power Inspectorate and the Swedish nuclear utilities. The main part of research and development work was carried out by ASEA-ATOM (design concepts) and Studsvik Energiteknik (coremelt phenomena).

A phase-1 progress report on FILTRA was issued in March 1981 (3). It indicated that the risk-reduction potential of vent-filter concepts is promising and that practical design solutions can be found. However, further experimental and modelling work was deemed necessary as a prerequisite for decision-making.

The Government did not wait for further research. In February 1981, they ordered a filter system to be installed in the Barsebäck plant by 1985. As a consequence of this decision, work on filtered venting accelerated and focused on implementation at the two 570 MWe BWR units at Barsebäck.

The FILTRA system for Barsebäck is shown in Figure 1. Briefly, pressure-relief ducts are connected to the the wetwell of each containment and are conducted to a large filter chamber, common to both units. The filter consists of a concrete structure filled with about 10,000 m^3 gravel.

Upon a core-melt accident resulting in containment overpressurization beyond design pressure, a rupture disk located close to the containment wall in the relief duct will break. Steam, gases, and arerosols will pass through the water pool of the wetwell which acts as a prefilter for iodine and some other fission products. Remaining amounts of the volatile products of the core melt will be discharged to the gravel bed. This unit acts as a condenser for steam and, according to large-scale experiments, will retain close to 100 percent of all fission products except the noble gases. The latter, however, will be delayed substantially before eventual release through the stack.

In their 1981 decision, the Government singled out Barsebäck for implementation of FILTRA because of its nearness to population centres (Malmo, Sweden and Copenhagen, Denmark). However, the Government also required that measures providing an equivalent level of safety be taken in the other 10 plants and that those measures be carried out before 1989.

Since 1981, the regulatory bodies (SKI/SSI) and the utilities have continued their joint research within the framework of the RAMA project (Reactor Accident Mitigation Analysis). The work aimed at improving the understanding of core melt phenomen a and was characterized by much interaction with similar work in other countries. The continued work established filtered venting of the containments as a major accident mitigating measure. It also provided a basis for a modified filter design involving a water pool of moderate size rather than the very voluminous gravel bed of the FILTRA project in Barsebäck. Aerosols and elemental iodine released upon severe fuel damage are absorbed in the pool water. The driving pressure from the containment forces the contaminated steam and gases to flow through a system of venturi nozzles located in the pool. The water droplet formation in the venturis greatly enhances absorption of particles. Thiosulfate addition to the pool water enhances iodine filtration. The entire filter volume is 300-400 cubic meter, including a gravel moisture separator located between the scrubber and the stack.

The containment venting and filtering projects are now implemented by the utilities (SSPB and OKG AB). The filter contracts were awarded to ASEA-ATOM and FLÄKT INDUSTRI who developed the specific design. The filters are called FILTRA-MVSS (Multi Venturi Scrubber System). Filters of the type described will be attached to the three PWR containments as well as to the seven BWR units.

The total upgrading of measures to mitigate the effects of severe core accidents in these reactors includes the following:-

- containment relief and filtering as described above

- improved reliability of the containment spray

- procedures for containment flooding (to the top of the core).

The Swedish authorities and utilities see the current round of modifications as the major step in severe accident backfits. The vent/filter systems in particular are an important safety feature for these plants. The utilities consider the nuclear power plants are safe, reliable and economical power producers which could technically continue operation beyond 2010.

The next section of this paper outlines the Swedish regulatory criteria and requirements for mitigation of severe accidents. Then, the applications to the BWRs and PWRs are described, including design bases for the equipment to be installed. A brief description of the Multi Venturi Scrubber System is given. Finally, the main conclusions of the current projects are summarized.

2
REGULATORY CRITERA AND REQUIREMENTS

The regulatory bodies involved in the licensing of nuclear power plants and radiation protection are the Swedish Nuclear Power Inspectorate (SKI) and the National Institute of Radiation Protection (SSI), respectively. In their task to interpret the Government decision on vent-filter systems, they arrived at the following general safety criteria:-

1. A degraded core accident shall not lead to any acute radiation fatality.

2. Emphasis is put on measures that prevent land contamination that would impede land usage.

3. The requirements with respect to limiting the release of radioactive substances should be equal for all plants regardless of site location.

4. The requirements arising from these criteria are not applicable to scenarios which are extremely improbable (such as a catastrophic rupture of the reactor pressure vessel).

These criteria were subsequently interpreted to mean that no more than 0.1 % of the core inventory of radioactive matter (excluding noble gases and organic compounds of iodine) be released.

The authorities in their comments to the general criteria in 1985 re-emphasized the importance of accident prevention, emergency preparedness and preservation of containment integrity. They pointed out the benefit of measures that ensure that containment over-pressure can be relieved and indicated that filling the containment with water to the top of the reactor core will improve containment cooling and can provide a means of core cooling and achievement of stable post-accident conditions. It was noted that a particular case exists for pressure suppression containments, in so far as a LOCA in combination with the loss of the suppression function will lead to rapid overpressurization. The relief of this overpressure direct to the atmosphere might not lead to excessive exposure, since the released steam should be fairly clean.

These guidelines and observations were partly based on analysis and alternative design approaches presented by the utilities.

The authorities also provided specific guidelines with respect to automation versus operator action, need for instrumentation to monitor release, and other tentative requirements. The measures finally decided upon should be completed by the end of 1988.

3 APPLICATIONS

In the analytical work done within the RAMA project it was soon discovered that accident mitigating measures would have to be specific to the individual plants. The utilities readily embarked on plant-specific studies which led to the designs now being implemented. The analysis was supported by comprehensive calculations using several containment response codes and source term codes. In particular, the MAAP program was used extensively (4).

3.1 Common design principles

Some design principles are common to all the ten plants considered, although they depart slightlyfrom those applied in the Barsebäck case.

All the plants will be equipped with pressure-relieving devices connected to MVSS filters. Oskarshamn 1 and 2 will share one filter unit; all the other units will have their individual filter, see figure 2.

The main design criteria include the following items:-

1) Maximum release of 0.1 % of core inventory (excl. noble gases and organic iodine compounds).

2) No operator action may be credited within 8 hours following an accident.

3) Protect against hydrogen explosion is required.

4) The single failure criterion for active components shall be fulfilled. A rupture disk is not considered to need redundancy. Other exceptions may materialize.

5) New structures and equipment shall be designed to maintain their safety functions during an earthquake corresponding to 0.15 g ground acceleration.

6) A defined set of accident sequences including loss of all AC power shall be covered for each plant. The defined sets should constitute envelopes over all credible scenarios.

7) Pressure relief of the containment shall be provided by automatic means as well as by manually initiated actions.

8) Means to fill the containment with water up to the top of the core shall be provided.

9) The reliability of the containment spray shall be upgraded.

10) Instrumentation to monitor important process and system parameters as well as the amount of radioactivity released shall be provided.

The principle of the pressure relief and filtering system is shown in figure 3. The vent line is opened automatically through a rupture disk upon containment overpressure (slightly above design pressure) and open isolation valves. The steam and non-condensing gases pass to the scrubber where heat is dissipated, steam is condensed and radioactive substances are absorbed. A parallel line can be opened by manual initiation for controlled pressure relief.

As mentioned above, the event sequences include loss of all AC power (blackout) including loss of the on-site diesel generator sets. In order to preserve the capability to inject spray water into the containment in this scenario, extra piping will be connected to the containment spray nozzle system. Water to the extra spray lines may be provided from the fire water system which can be driven by on-site pumps as well as by the fire brigade. The latter alternative cannot however be credited until some time after the accident has occurred.

The spray system should thus be reliable enough to ensure containment spray which is essential for the following purposes:-

- to reduce containment pressure

- to cool the containment atmosphere

- to wash out radioactive substances from the containment atmosphere

- to fill water into the containment up to the top of the core.

The spray function is associated with a decontamination factor of at least 10 with respect to airborne radioactivity.

3.2 Pressurized water reactors (PWR)

There are three PWR units at Ringhals with large dry containments (figure 4). The design principles given in the preceeding section cover most of the items specific for these PWRs.

The accident sequence that the arrangements are designed to mitigate is loss of all AC-power and loss of the turbine driven auxiliary feedwater pump. This will be an envelope covering most other possible sequences. Such sequences, which are subject to further study, include loss of reactor shutdown and those involving stuck-open relief valves.

Figure 5 illustrates containment pressure history in the blackout sequence with containment spray start at two different points in time.

3.3 Boiling water reactors (BWR)

There are three BWR units at Oskarshamn, the first two of which will share one MVSS filter, and the third unit will be equipped with its own filter unit. There is one BWR at Ringhals and three units at Forsmark - all with individual filters. The Ringhals 1 and Oskarshamn 1 and 2 BWRs have external recirculation loops and the other units have internal pumps.

All units have pressure suppression containments similar to the U.S. Mark II type. In the older units with external pumps (figure 6) the pressure suppression pool covers the entire bottom area of the containment, whereas the pool forms an annular region in the internal pump plants (figure 7).

The BWR accident sequences considered to cover credible scenarios include

- loss of all AC-power (blackout)

- LOCA in combination with degraded pressure suppression function.

The blackout sequence leads to core melt. In order to protect the floor of the lower drywell and containment penetrations in this area in the internal pump plants, arrangements are made to fill this space with water from the suppression pool. The transfer of water may be initiated automatically or by manual means upon information available to the operator, indicating that a severe accident may be imminent.

The LOCA and loss of pressure suppression sequence (AD in the WASH 1400 denotation) may - if no countermeasure is taken - precede a core melt, since the containment integrity is jeopardized. The approach pursued in this case involves the assumption that pressure suppression is degraded due to a leak in the drywell-wetwell boundary. The maximum leak considered corresponds to the circumferential break of one downcomer pipe to the suppression pool. The postulated LOCA then leads to rapid overpressurization. This event is counteracted by furnishing a wide (600 mm diameter) relief channel from the top of the containment equipped with a rupture disk (figure 8). Since the LOCA is assumed to start from normal operating conditions, the steam to be relieved is not substantially contaminated. Therefore, this containment rupture disk is allowed to open direct to the outside atmosphere. It will open at a fairly high pressure (above design pressure) in order not to be actuated inadvertently. After pressure relief, the line is reclosed by valves in the open pipe. The resulting dose to persons off-site is calculated not to exceed the annual dose permissible during normal plant operation.

ATWS-sequences sequences are not included in the scenarios considered, since dual function control rod insertion, pump runback, and manually initiated boron injection systems render the reactor shutdown function extremely reliable.

Figure 9 illustrates a typical containment pressure history following the blackout sequence.

4 FILTER DESIGN

The FILTRA-MVSS concept (Multi Venturi Scrubber System) is designed and optimized into a compact unit which may be located in close vicinity of each reactor containment. In order to serve as a protection device against overpressurization of the containment a suitable pressure relief system also is included. The concept has been designed by ASEA-ATOM and FLÄKT INDUSTRI who developed the specific scrubber system.

The FILTRA-MVSS concept comprises the following main process steps:

- o A pressure relief system
- o A venturi scrubber system
- o A pool for iodine separation
- o A moisture separation system
- o A service system
- o A concrete pressure vessel

The pressure relief system is directly connected to the multi venturi scrubber system. The gas and/or the steam from the containment is directed to the distribution chamber inside the pressure vessel. From this chamber the medium is distributed into a number of venturi nozzles (see figure 10).

The distribution system passively and automatically engages the relevant number of venturi nozzles in correspondence with the actual pressure in the reactor containment. The whole system is designd to be fully selfcontrolled and independent of any type of active control unit or auxiliary power.

The FILTRA-MVSS building is made of concrete. The inside of the pressure vessel is provided with a stainless steel liner. The concrete vessel could either be designed as a conventional reinforced concrete vessel or in the form of a prestressed concrete vessel depending on the relevant design criteria.

The pressure relief system is located in a room in the upper part of the building. This room will contain both the equipment for the pressure relief system and the moisture separation system.

The FILTRA-MVSS unit can also be combined with a number of back-up functions, e.g. to drain the vessel and to fill it up with water, to heat or cool the poolwater and to take out samples of the pool water. All the valves for such functions will have to be operated from behind a radiation protection shield.

The FILTRA-MVSS concept can accommodate also the effects of hydrogen combustion without any major design alterations.

The venturi nozzle provides a key mechanism for the removal of particles and elemental iodine. Water is sucked into the gas stream at the nozzles, and water droplets of suitable size and velocity are formed. The water/particle interaction is grately amplified by this design, and several physical phenomena contribute to a high absorption rate. Iodine retention is further increased by adding thiosulfate solution to the pool water.

Since a total decontamination factor of 1000 is required for the source term, and the containment spray function (cf section 3) accounts for a factor 10, the multi venturi scrubber should be designed to yield a decontamination factor of 100. In the PWR case, however, there are some sequences in which the containment spray may be unavailable for a few hours. This circumstance is compensated for by requiring a MVSS decontamination factor of 500 in the PWR applications.

5 CONCLUSIONS

The new Swedish regulations effectively amount to consider degraded core accidents - including a core melt - to be design basis events. The measures now implemented in all Swedish nuclear power plants are designed to mitigate the consequences of these accidents in such a way that no acute fatality and no severe land contamination would occur. The main mitigating measures consist of venting containment overpressure through filters, improved reliability of containment spray, and procedures for containment flooding to the top of the reactor core.

6 REFERENCES

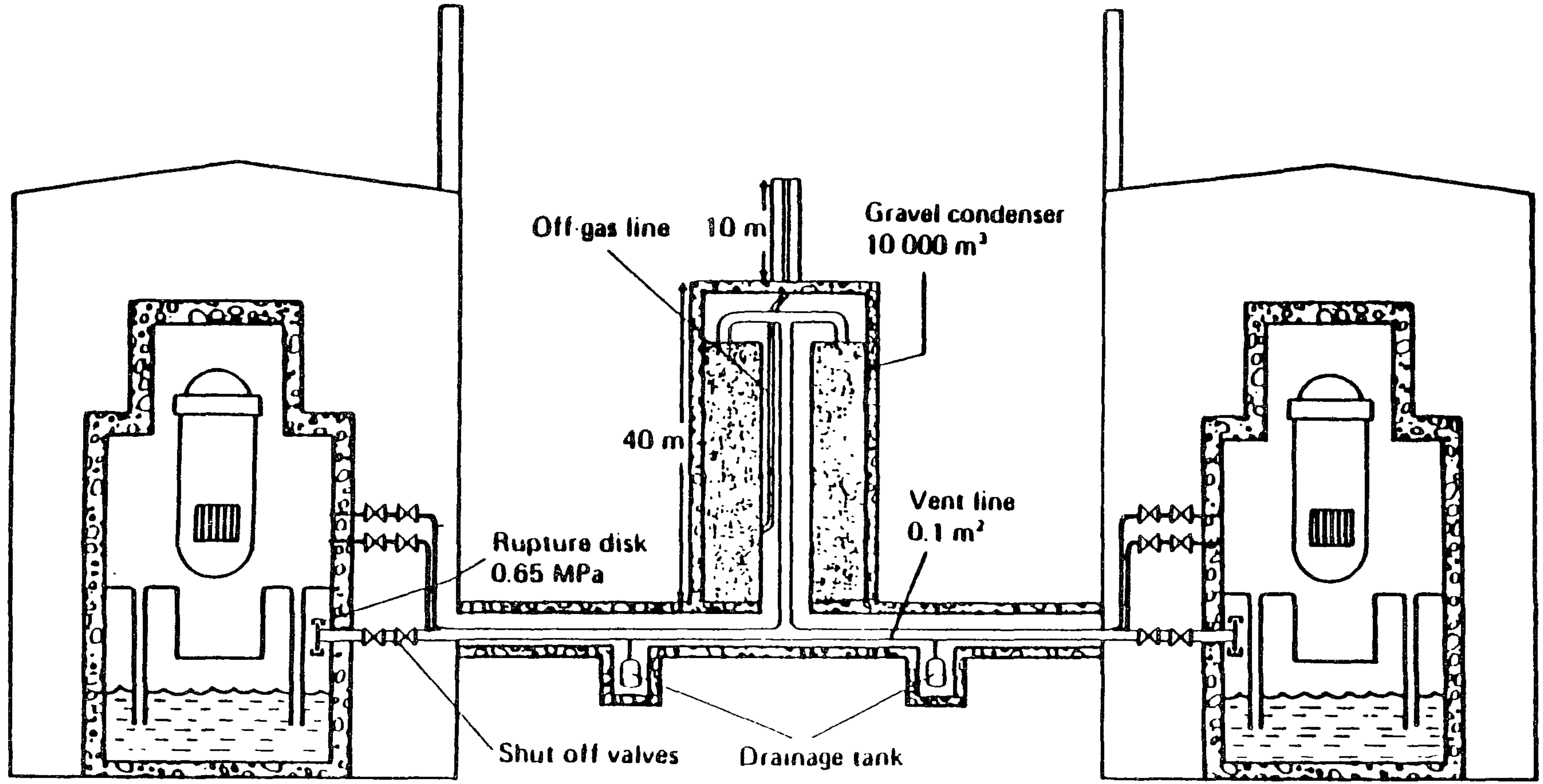
1. Swedish Reactor Safety Commission, "Safe Nuclear Power?", Stockholm (1979), SOU 1979:86.

2. L I Tirén, "Safety Considerations for Light Water Reactor Nuclear Power Plants: A Swedish Perspective", Institute for Energy Analysis, Oak Ridge Associated Universities, 1983. DE83015638, ORAU/IEA-83-7 (M).

3. K Johansson et al., "Design Considerations for Implementing a Containment Vent-Filter Plant at Barsebäck, Sweden", Internat. Meeting on Thermal Nucl. Reactor Safety, Chicago, Aug 29 - Sept 2, 1982, Session 30 (Log no 255).

4. Fauske & Associates, "MAAP-Modular Accident Analysis Program. User's Manual Vol I & II", IDCOR Techn. Report 16.2-3 (1983).

ASEA-ATOM



BARSEBÄCK NUCLEAR POWER PLANT (SWEDEN)
SCHEMATIC DIAGRAM OF CONTAINMENT VENT/FILTER SYSTEM (FILTRA)

FIGURE 1

FILTRA - MVSS

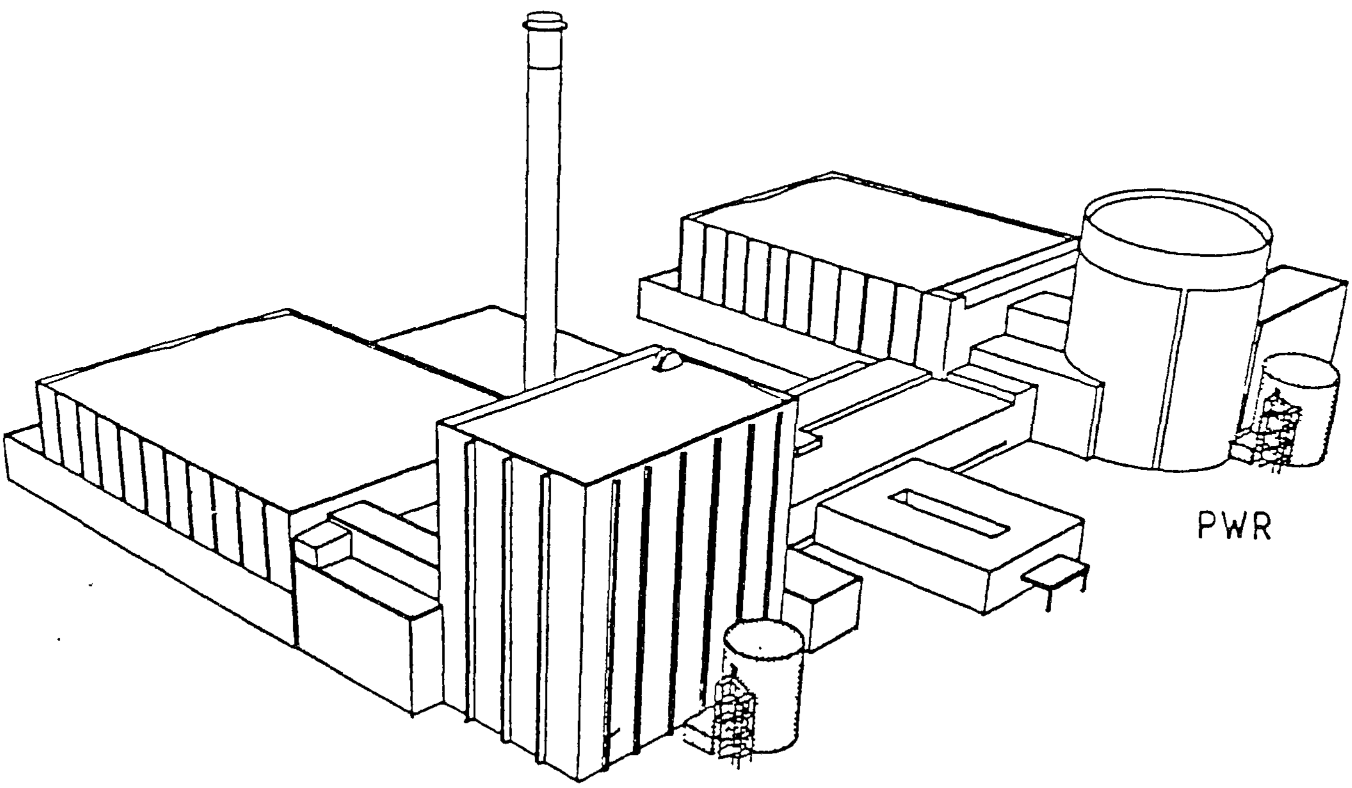


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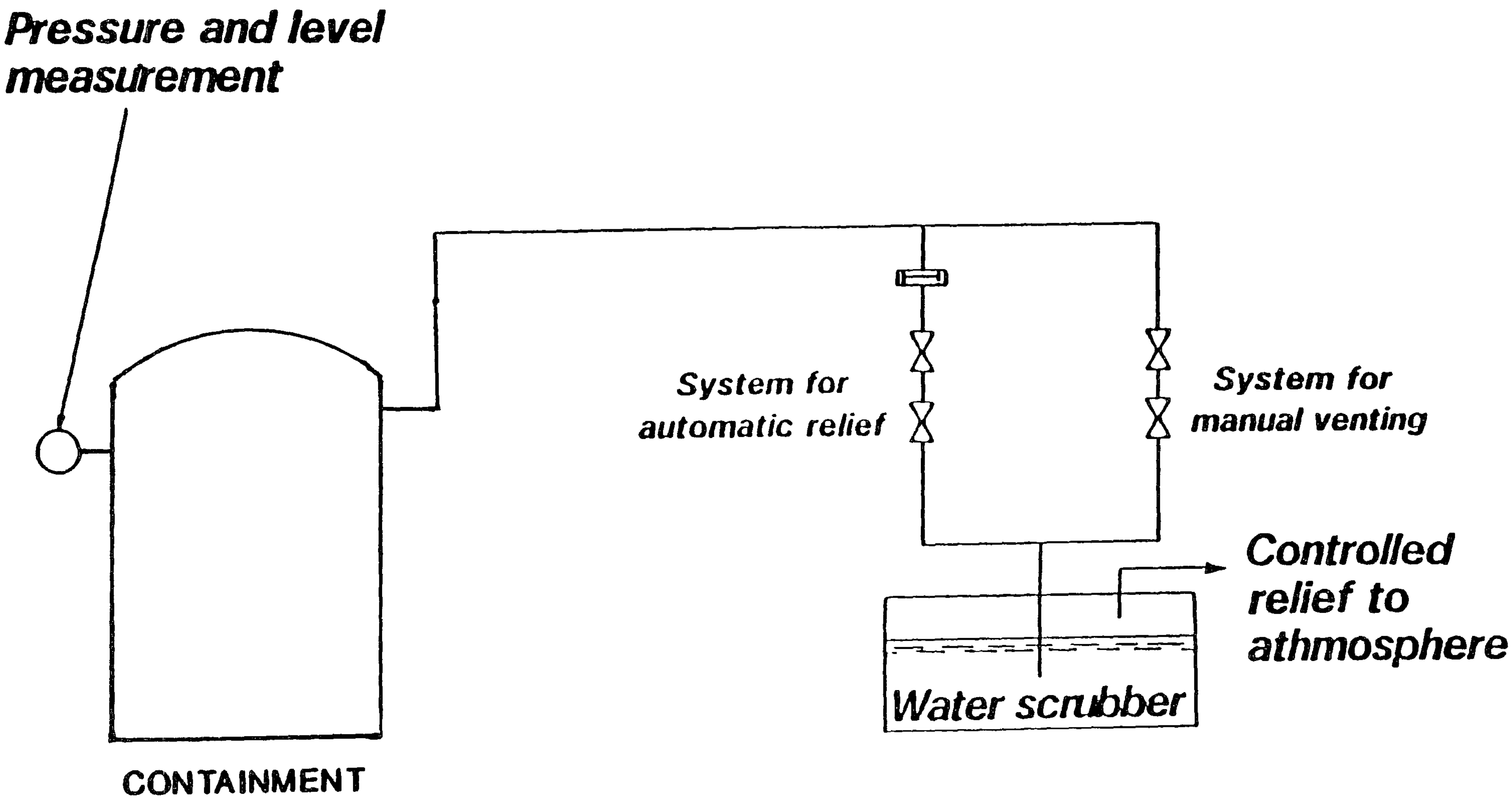
FIGURE 2

ASEA-ATOM



SCHEMATIC OF CONTAINMENT VENT/FILTER SYSTEM

FIGURE 3

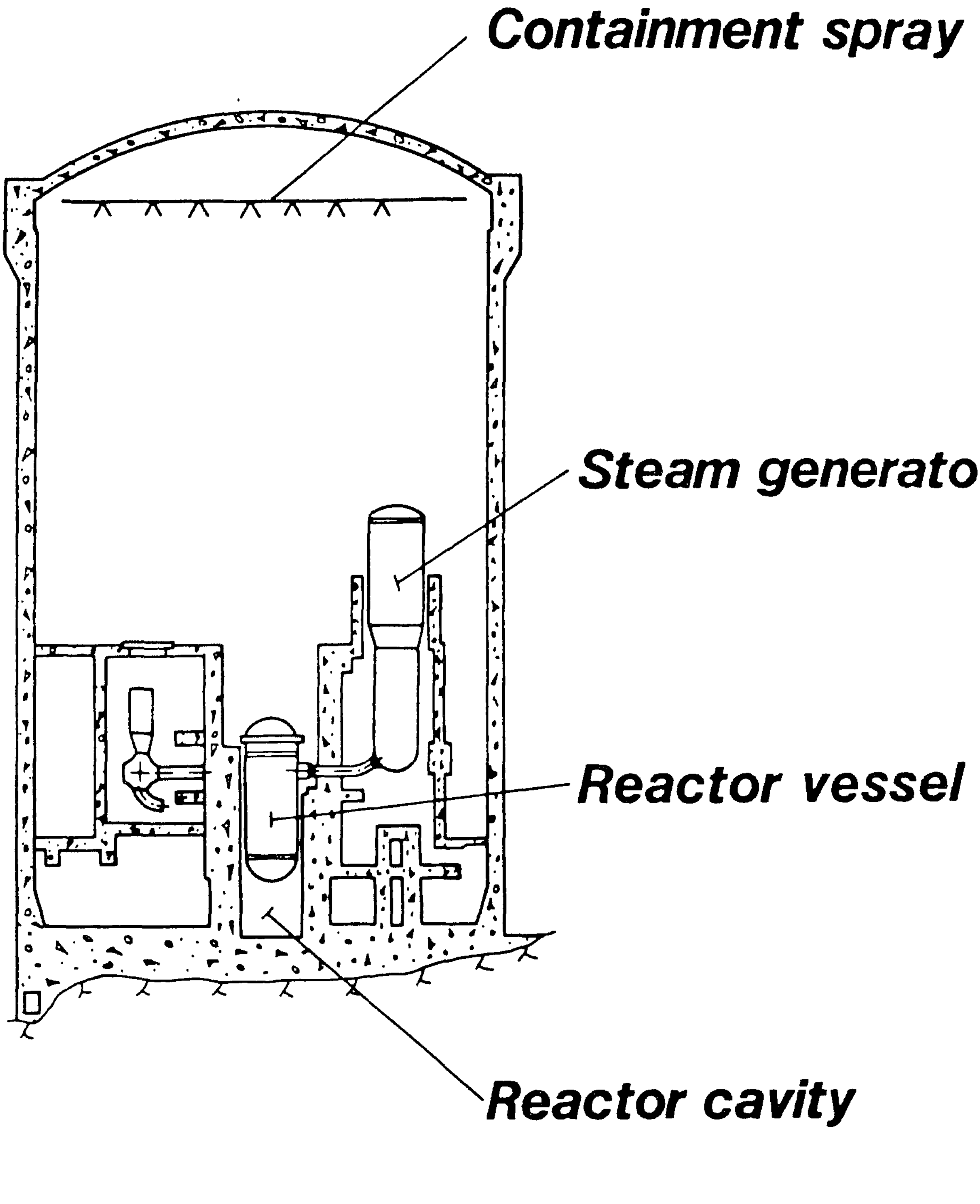
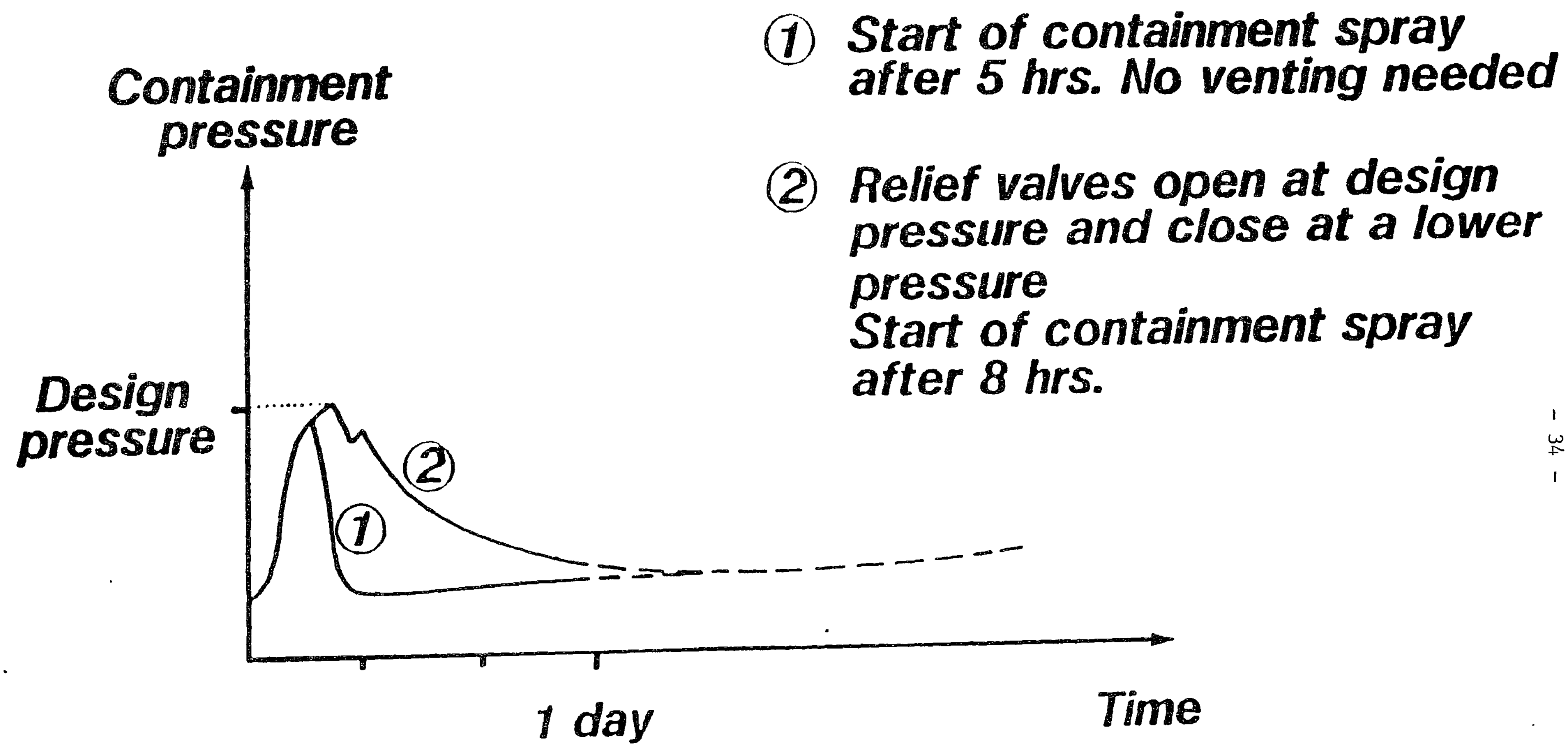


Fig. 4 RINGHALS 2
CONTAINMENT CONFIGURATION



TYPICAL PRESSURE HISTORY FOR SEVERE ACCIDENT SCENARIO
IN A SWEDISH PWR – RINGHALS 2–4

FIGURE 5

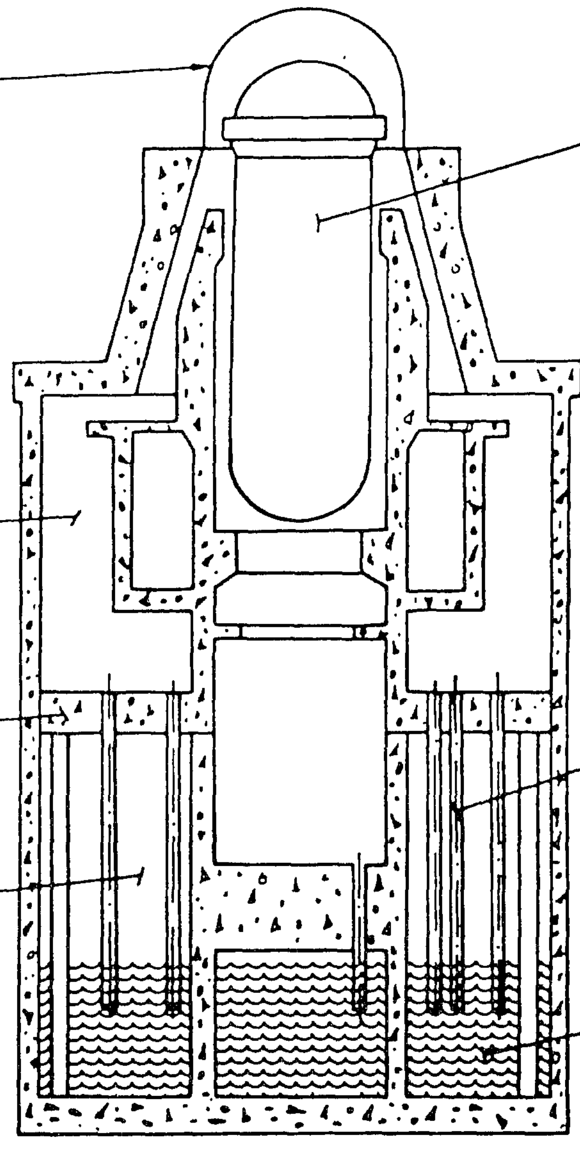


Fig. 6 **RINGHALS 1 CONTAINMENT CONFIGURATION**

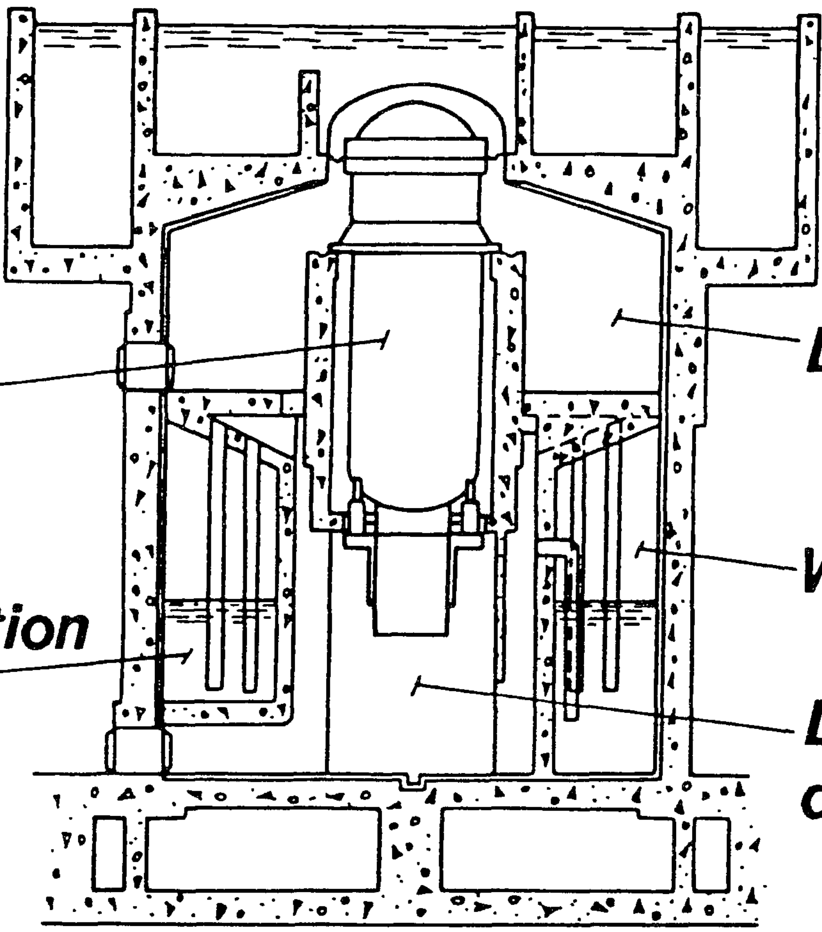
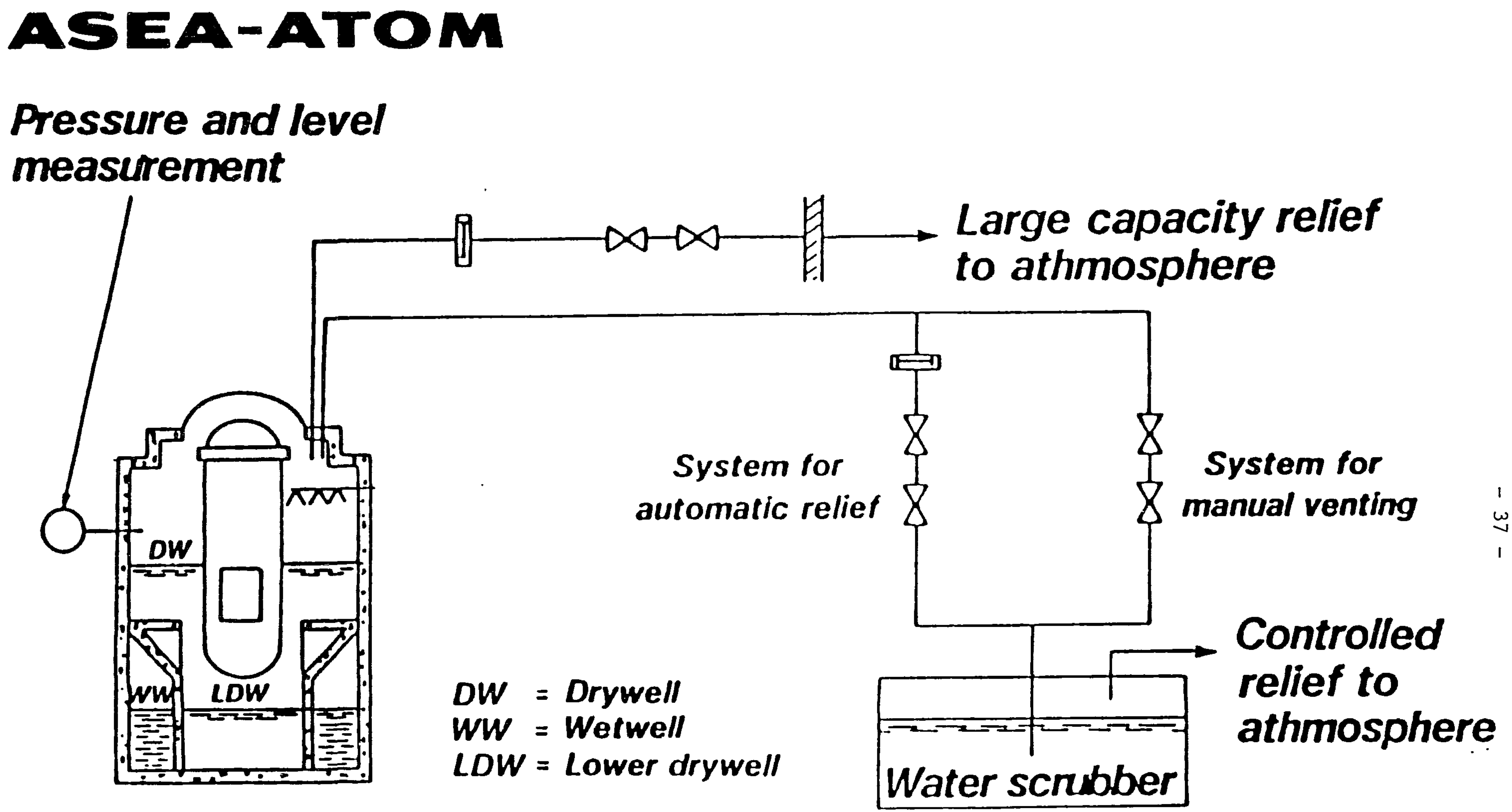
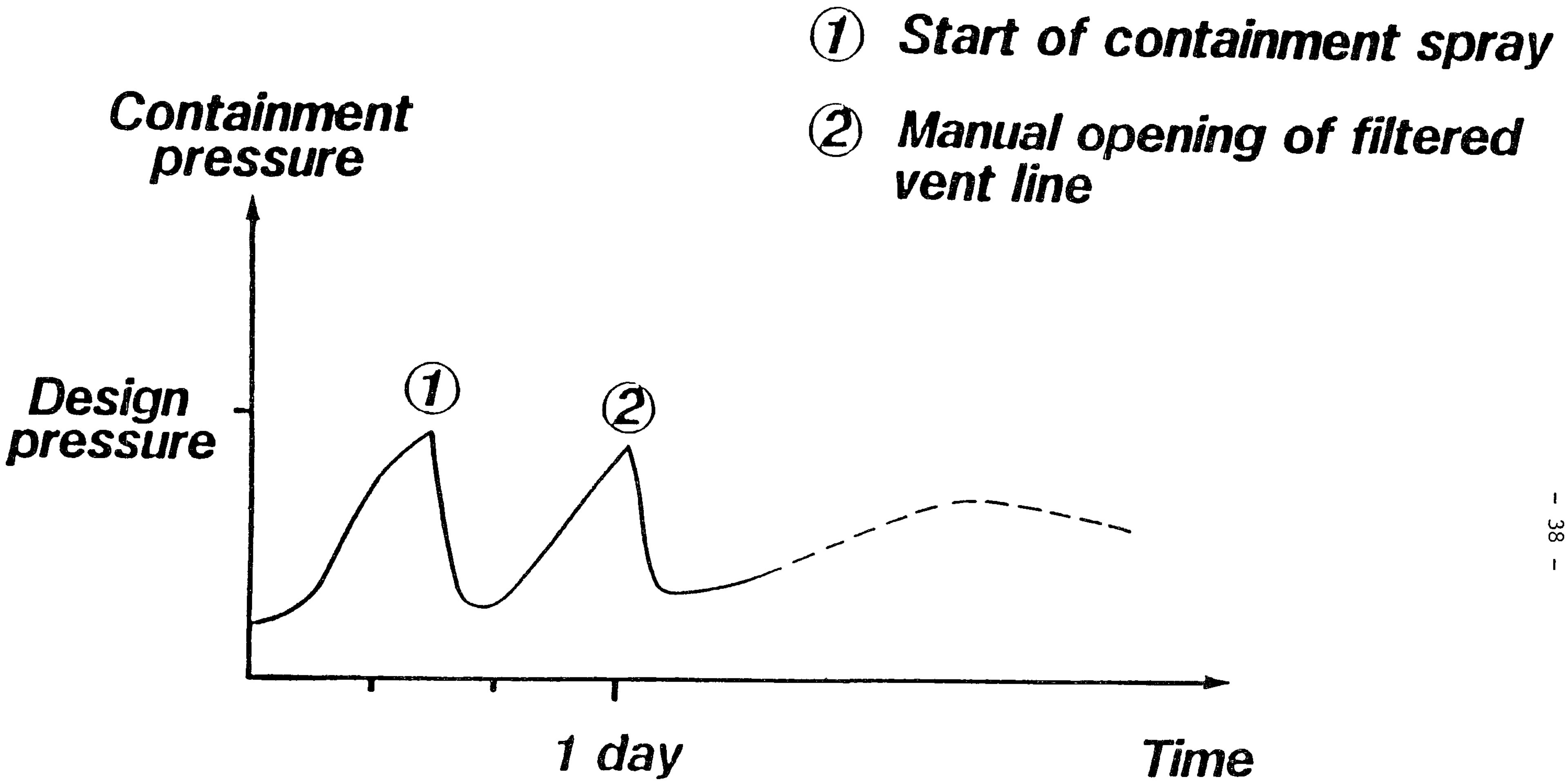


Fig. 7 FORSMARK 3
CONTAINMENT CONFIGURATION



SCHEMATIC OF CONTAINMENT VENT/FILTER SYSTEM
TO BE USED AT RINGHALS 1 AND FORSMARK 1–3

FIGURE 8



TYPICAL PRESSURE HISTORY FOR SEVERE ACCIDENT SCENARIO
IN A SWEDISH BWR - RINGHALS 1 OR FORSMARK 1-3

FIGURE 9

FILTRA - MVSS

FILTERED CONTAINMENT VENTING
WITH
MULTI VENTURI SCRUBBER SYSTEM

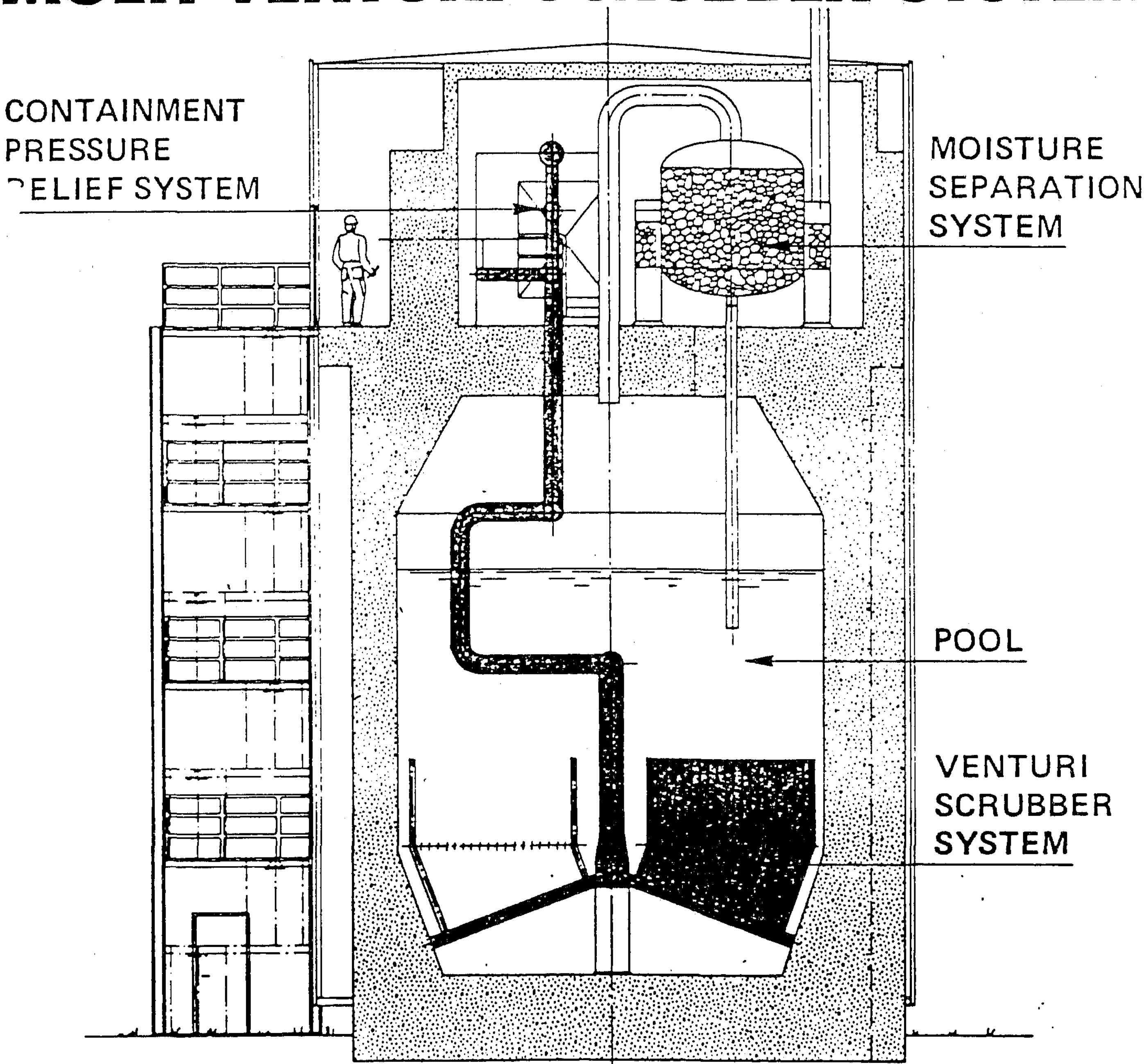


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FIGURE 10

METHOD OF INVESTIGATING ACCIDENT PROCESSES AT NUCLEAR POWER PLANTS (NPP)

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At present studies of accident processes are increasingly done by means of computer simulation. Complexity of mathematical description of spatially conjugated neutron-physical processes, heat exchange, hydrodynamics taking into account possible destruction of individual elements makes it impossible to solve these tasks in general for NPP even with the help of modern technical means.

Therefore increasingly applied are studies of accident processes of individual elements of NPP. In this context initial and boundary conditions, generally, are assumed to be independent of operation of a given element in a circuit system [1-5].

The present report suggests a method of investigating accidents at NPP which is based on dividing the process by separate stages. The results obtained at each stage represent initial and boundary conditions for the successive stage. For each stage specific mathematical models are built.

Fig.1 shows stages studies of NPP accident processes. Mathematical modeling is based on a generalized transport equation which in non-dimensional form looks as follows:

$$\frac{\partial c}{\partial \tau} = -div\, c\, \bar{w}_s + K_{вн} J_{вн} + K J - K_м\, div\, J_м ; \qquad (I)$$

where C - substance, generalized characteristic, which defines energy, density, momentum etc.

KBH, K, Km - corresponding scaling criteria;

Ws - relative velocity of restricting surface;

JBH - relative specific volumetric power of external source;

J - relative specific volumetric power of internal source;

JH - relative flow of substances owing to molecular processes.

At the first stage a mode of possible operation of NPP is being determined. A mode of possible operation - is a range of NPP operating regimes, where parameters do not exceed permissible limits. For this given mode one should determine optimal in a particular sence regimes of NPP operation.

To this end a universal method of investigation has been developed for a wide range of tasks. The method is based on an assumption that any installation works as a system consisting of the following standard elements: reactor, turbines, regenerators, condensers,mixers, pumps and a bulk of termo-physical properties of coolant. By selecting the above mentioned elements one can obtain practically any technological diagram of a facility. Characteristics of elements are described mathematically in such a way that they do not depend on a specific type of elements,which allows their multiple use while calculating variable operating regimes. This approach to the description of installation permitted to develop an algorithm of a code for digital computer [6].

For calculation the following initial data is needed:

1) technological diagram and the sequence of elements' connections;

2) pressure, temperature, coolant flowrate relevant to calculated regime, which is obtained from a calculation of thermodynamic cycle;

3) surfaces and geometry of reactor, regenerator and condenser;

4) values of through put among units for variable operating regimes.

As a result of a facility's calculation for variable operating regimes it is possible to obtain the following information for different modes of operation:

1) effective coefficients, which are the initial data for determination of power efficiency of a facility;

2) temperatures, pressures and coolant flowrate in all units of a facility;

3) temperature of fuel element cladding along the channel;

5) reactivity variations as a result of temperature and density effects;

6) level of coolant superheating at the regenerator outlet;

7) level of coolant dryness at the turbine inlet;

8) level of liquid coolant overcooling at the pump inlet.

On the basis of variable operating regimes obtained different programmes of a possible operating mode is selected. Fig. 2 shows a possible operating mode of NPP with dissociating coolant. Technological diagram of such NPP is also shown on the figure. From the condenser liquid coolant is supplied to a full flow regenerator where it is evaporated and superheated and then fed to a reactor. From the reactor coolant flows to a high pressure turbine, regenerator, low pressure turbine and to a condenser. Then the thermodynamic cycle becomes closed. Minimal level of steam superheating at the reactor inlet, maximum coolant pressure and temperature at the reactor outlet, are considered as limiting factors.

The operating regime which provides maximum value of the power efficiency is considered to be optimal regime. This factor represents a ratio between useful power and consumed power for a specific period of time and takes into account the operation time of NPP in different regimes.

$$\eta_{эн} = \frac{\int_0^T N_п \, d\tau}{\int_0^T \frac{N_п}{\eta} \, d\tau} . \qquad (2)$$

Thus, the investigation on the first stage should result in defining initial conditions for initial state before an accident and in assessing margins for limiting parameters.

At the second stage the dynamics of accident process in the NPP circuit are studied taking into account operation control and instrumentation system. For this purpose a one- dimensional mathematical model of a fuel channel is used. The need to take into account the effects of the elements conjugated with coolant is met by the criterion:

$$\beta = \frac{C_э M \Delta t_э}{\tau_э Q}; \tag{3}$$

which defines heat fraction, taken by the channel. In this equation C - thermal capacity of the element, M - its mass, Ste - temperature variation, T - time constant, Q - thermal power of the unit. Mathematical model of the second stage is presented in the following way. Main equations of this model look like:

$$\rho C_p \frac{\partial T}{\partial \tau} + \rho C_p w \frac{\partial T}{\partial x} - \left(\frac{P}{\rho C_p T}\right)_0 \left(\frac{\partial P}{\partial \tau} + w \frac{\partial P}{\partial x}\right) = K_q \dot{q} + \left(\frac{\ell J_{вн} h_{вн}}{w \rho C_p T}\right)_0 J_{вн} h_{вн} - \left(\frac{\ell J_{вн} h}{w \rho C_p T}\right)_0 h J_{вн}; \tag{4}$$

$$\rho \frac{\partial w}{\partial \tau} + \rho w \frac{\partial w}{\partial x} + Eu \frac{\partial P}{\partial x} + \xi \frac{\rho w^2}{2 \frac{d_э}{\ell}} = \left(\frac{\ell J_{вн} w_{вн}}{\rho w^2}\right)_0 J_{вн} w_{вн} - \left(\frac{\ell J_{вн}}{\rho w}\right) J_{вн} w; \tag{5}$$

$$F \frac{\partial \rho}{\partial \tau} + \rho \frac{\partial F}{\partial \tau} + \rho w \frac{\partial F}{\partial x} + F \frac{\partial \rho w}{\partial x} = \mp F K_{вн} J_{вн}; \tag{6}$$

$$C_{ст} \rho_{ст} \frac{\partial T_{ст}}{\partial \tau} = \left(\frac{q_{ст} \tau}{C_{ст} \rho_{ст} T_{ст}}\right)_0 q_{ст} - \left(\frac{q \tau}{C_{ст} \rho_{ст} T_{ст}}\right) q. \tag{7}$$

The mathematical model gives a possibility to conduct studies on leakages at various points of the technological diagram, on loss of power, failures of control and instrumention and other accident situations.

Analysis of β criterion for different elements shows that vessels, conchoidal tubes, moderate length pipelines do not have significant influence on the coolant temperature. Moreover with the increase of NPP regimes of operation their influence diminishes.As a result of these studies we obtain changes in the parameters in the accident unit for example in the reactor, boundary conditions by coolant parameters for the accident unit owing to the influence of NPP circuit and control and instrumentation, boundary conditions for branch pipes, covers, vessels and other components. The results obtained may have influence on the changes in the programme of variable regimes and initial condition i.e. on the results of the first stage.

On the third stage dynamics of a unit and of individual components is studied in detail. For the unit more detailed spatial mathematical models is being used which takes into account the interaction of channels and temperature fields of fuel elements. The method of finite elements is used to determine the temperature fields in branch pipes, vessels, covers and other elements of any configuration. Density, heat conductivity coefficient, thermal capacity, internal source and temperature are assumed to have mean values in volume [7].

The results obtained may clarify the assumptions with respect to accident unit and elements which were made while modeling NPP, i.e. clarify the results of the second stage. On the basis of proposed method the following data is obtained:

1. Dynamics of accident processes development.
2. The most vulnarable places in accident units and elements.
3. Time of reaching critical values of parameters.
4. Parameters on which an emergency protection of the reactor should be based.
5. Guidelines on the programmes of regulation, units' design and materials.
6. Emergency protection algorithms.

The data obtained represent also the initial conditions for investigating accident process causing destruction.

Fig.3 shows, as an illustration, the results obtained with respect of the accident with 100% load loss at the turbogenerator. As is seen from the given data the facility turns to a new equilibrium regime, while pressure in the condensor increases due to additional heat dumping. Fig.4 shows the results on dynamics of changes in pressure, coolant flow rate and temperature at the reactor inlet during NPP complete loss of power. In this case pump's runout has not been taken into account. After 6 sec. coolant flowrate still maintained at 50% of the nominal owing to coolant accumulation.

Bearing in mind that on the one hand, results of investigating accident regimes have major effect on safety and, on the other hand, these studies are based in various countries on different methods it would be advisable to develop uniform international guidance for NPP accident regime studies which would determine necessary scope of accident processes studies, recommended mathematical models, methods and programmes, common physical and thermophysical properties of NPP coolants and materials.

REFERENCES

1. Nucl. Technol., 1981, 54, No. 3, 398-409.
2. Nucl. Eng. and Des., 1982, 71, No. 1, 33-43.
3. Nucl. Technol., 1983, 60, No. 2, 278-290.
4. Trans. Amer. Nucl. Soc., 1984, 47, 314-315.
5. Nucl. Eng. and Des., 1985, 85, No. 1, 71-82.
6. SHAROVAROV, G.A., Dynamics of nuclear power plants with dissociating coolant. - Minsk: Nauka i tekhnika, 1980. - 239 pp.
7. SHAROVAROV, G.A., Physics of non-steady-state processes in nuclear power plants. - Minsk: Nauka i tekhnika, 1985. - 206 pp.

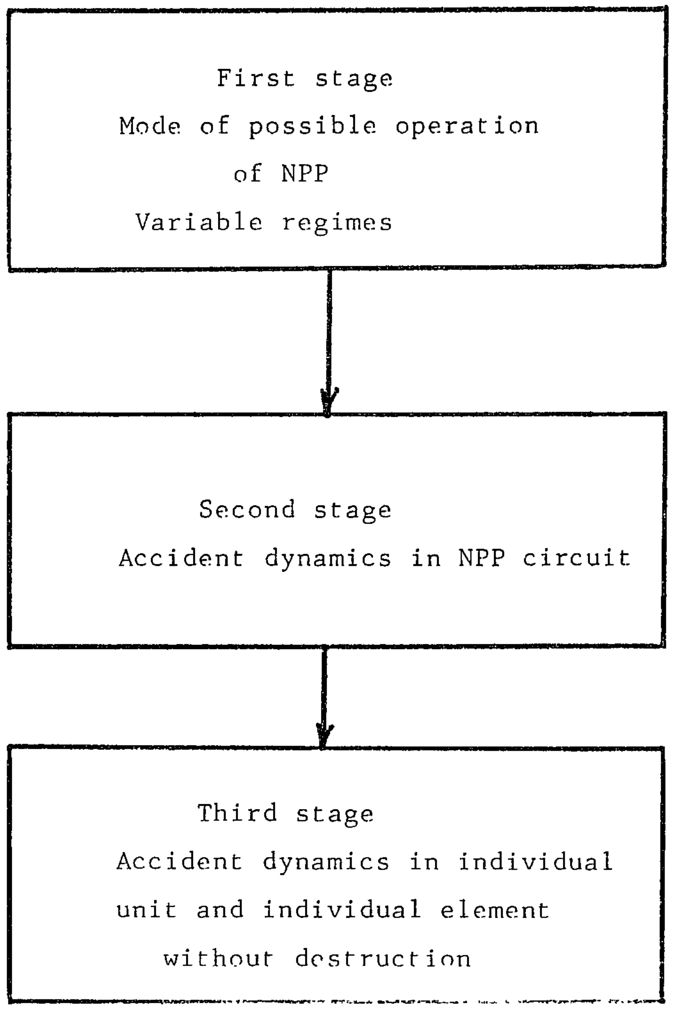

Fig. 1. Stages in the studies of NPP accident processes.

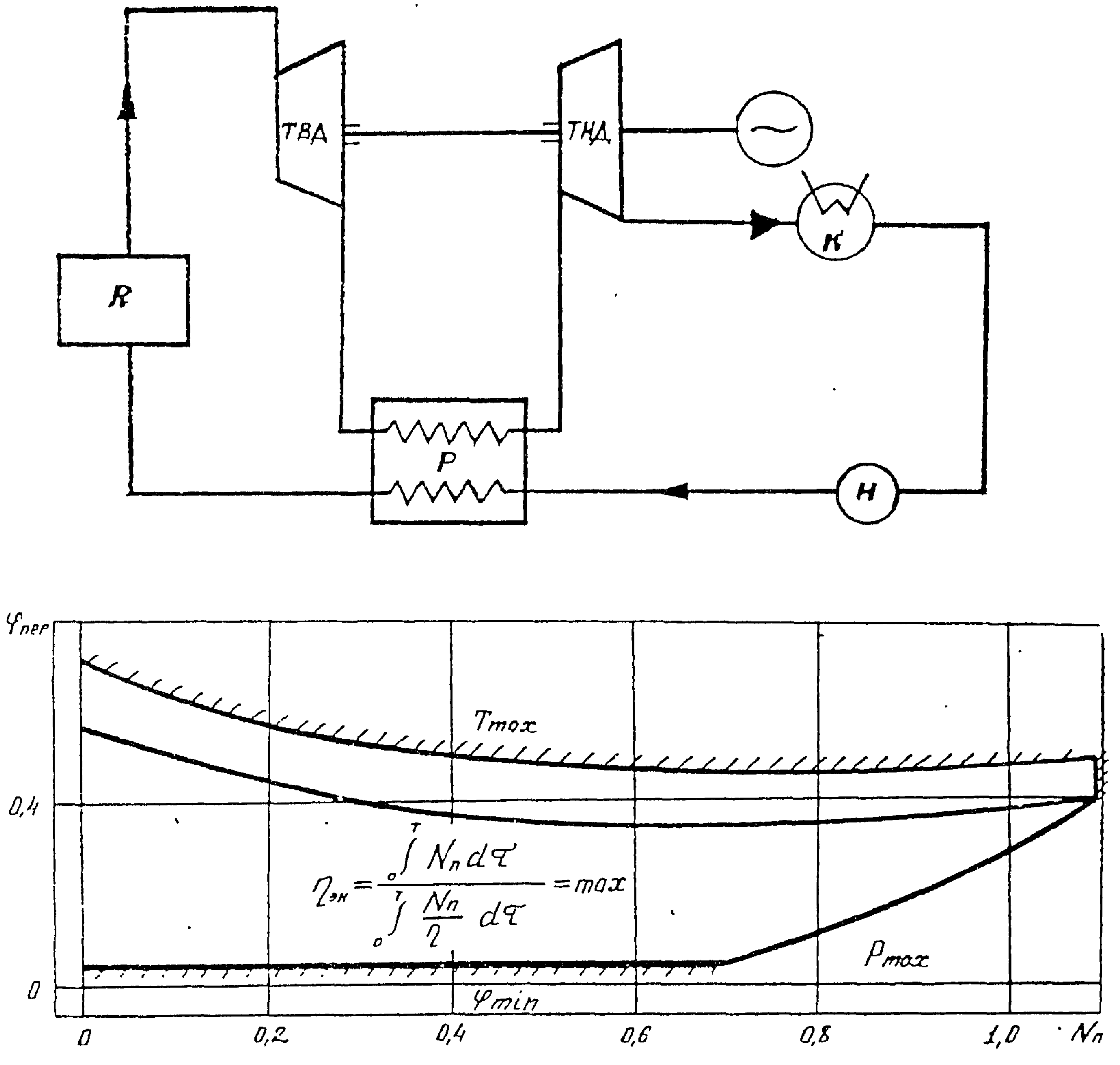

Fig.2. Technological diagram and mode of possible operation of NPP.

$$\rho C_p \frac{\partial T}{\partial \tau} + \rho C_p W \frac{\partial T}{\partial x} - \left(\frac{P}{\rho C_p T}\right)_0 \left(\frac{\partial P}{\partial \tau} + W \frac{\partial P}{\partial x}\right) = \tag{1}$$

$$= K_q \frac{Q}{V} + \left(\frac{\ell J_{вн} h_{вн}}{W \rho C_p T}\right)_0 J_{вн} h_{вн} - \left(\frac{\ell J_{вн} h}{W \rho C_p T}\right)_0 h J_{вн};$$

$$\rho \frac{\partial W}{\partial \tau} + \rho W \frac{\partial W}{\partial x} + \varepsilon_i \frac{\partial P}{\partial x} + \xi \frac{\rho W^2}{2 \frac{d_э}{\ell}} = \tag{2}$$

$$= \left(\frac{\ell J_{вн} W_{вн}}{\rho W^2}\right)_0 J_{вн} W_{вн} - \left(\frac{\ell J_{вн}}{\rho W}\right)_0 J_{вн} W;$$

$$F \frac{\partial \rho}{\partial \tau} + \rho \frac{\partial F}{\partial \tau} + \rho W \frac{\partial F}{\partial x} + F \frac{\partial \rho W}{\partial x} = F K_{вн} J_{вн}; \tag{3}$$

$$C_{ст} \rho_{ст} \frac{\partial T_{ст}}{\partial \tau} = \left(\frac{q_{ст} \tau}{C_{ст} \rho_{ст} T_{ст}}\right)_0 q_{ст} - \left(\frac{q \tau}{C_{ст} \rho_{ст} T_{ст}}\right) q; \tag{4}$$

$$\frac{\partial n}{\partial \tau} = \frac{\rho^x \beta}{\ell} n \psi(x) + \sum_{i=1}^{6} \lambda_i C_i; \tag{5}$$

$$\frac{\partial C_i}{\partial \tau} = \frac{\beta_i}{\ell} n \psi(x) + \lambda_i C_i; \tag{6}$$

$$q_T(\tau, x) = N_0 n + A \cdot F_0 \left\{ \sum_{i=1}^{15} A_i \exp(\lambda_i \tau) - \right. \tag{7}$$

$$\left. - \sum_{i=1}^{15} A_i \exp\left[\lambda_i (\tau + T_{обл})\right] \right\}$$

Fig.2a. Main equations of the mathematical models of a given stage.

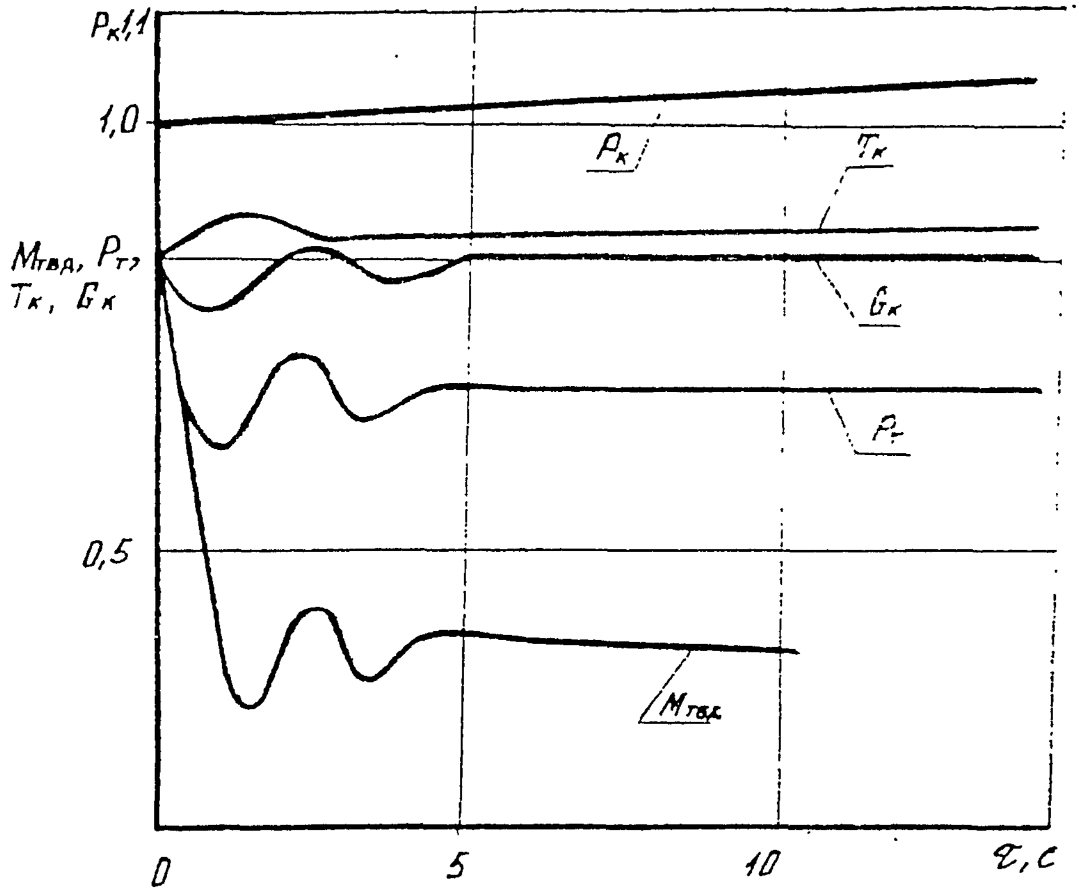

Fig.3. Changes in main NPP parameters during loss of load at the turbogenerator.

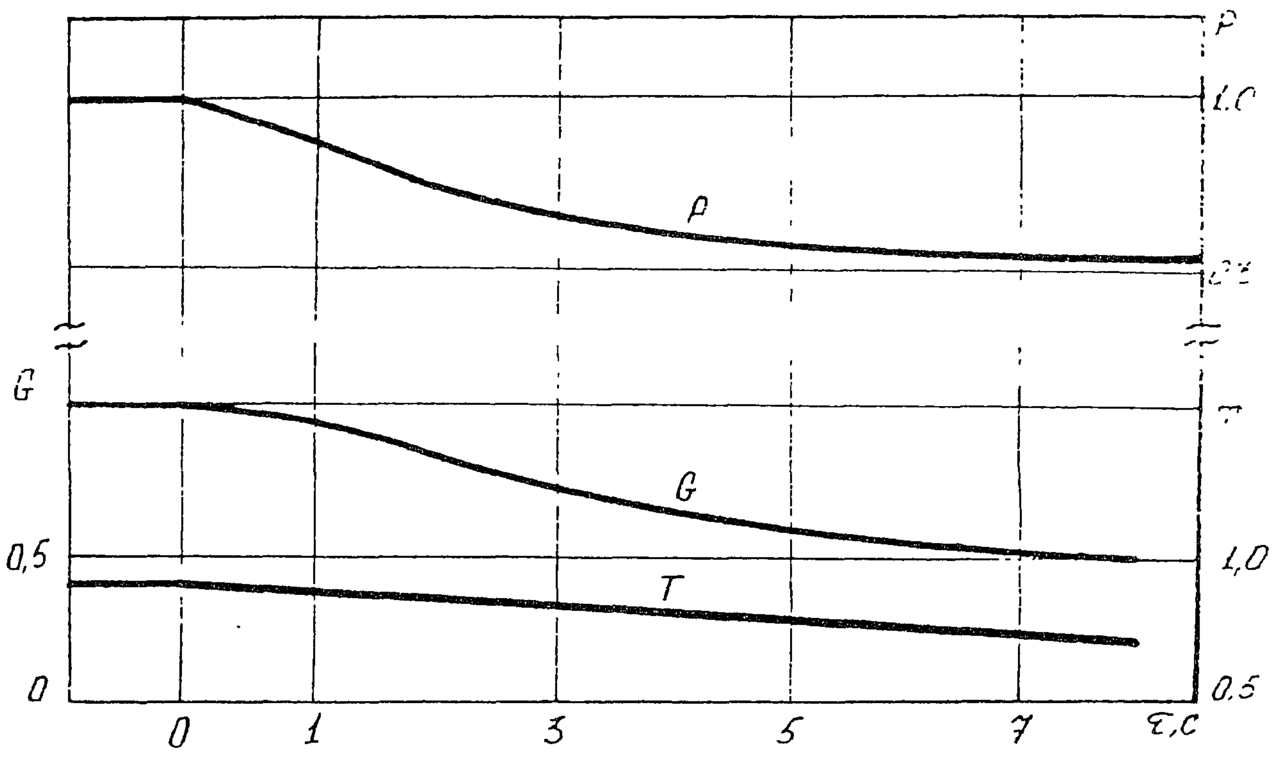

Fig.4. Changes in coolant flow rate, pressure and temperature at the reactor inlet during NPP loss of power.

ACCIDENT PROCESS DYNAMICS AT NUCLEAR POWER PLANTS WITH DISSOCIATING COOLANT

V.B. Nestorenko, G.A. Sharovarov, A.G. Shashkov

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NPP with dissociating coolant has in comparison with other NPP's a number of substantial differences affecting accident process dynamics. Above all they are related to its technological layout and thermodynamic cycle. The technological layout is presented in Fig. 1. Liquid coolant is fed from a condenser into a full flow regenerator and then is supplied as a steam into reactor where additional superheating takes place. The superheated steam feeds high-pressure turbine and then the regenerator where heat transfer to cold coolant occurs. Steam from the regenerator is fed to the low-pressure turbine and them to the condenser. The main feature of this technological layout is a combination of gas-cooled reactor and gas-liquid thermodynamic cycle in a single-loop plant. Such combination is impossible in case of using water steam since its vapourization requires much heat and full-flow regeneration is impossible.

The use of gas-liquid cycle results in generation of large amounts of liquid coolant in a technological loop with maximum pressure in cycle. It creates favourable prerequisites for maintaining continuous circulation through the core in case of electric cutoff mode of NPP due to passive flow of coolant from high pressure volume into low-pressure volume, i.e. condenser. This feature is essential for gas-cooled reactors since heat accumulating capacity of coolant in core is small and time constant is of the order of fractions of second. One should also mention low values of fuel rod time constants (1,5 - 2 s). Therefore the main task for providing safety in NPP with gas-cooled reactors is to maintain continuous coolant circulation through the core.

In simulating accident processes a question of applicability of static heat transfer coefficients for calculating dynamic processes arises. Based on theoretical and experimental research it was shown that for the majority of non-stationary processes with dissociating coolant the dynamic error does not exceed 10-15%. Fig. 2 shows dependence of non-stationary heat transfer coefficient on the rate of fuel rod cladding temperature change for ideal and dissociating coolant.

To model turbines in the NPP loop the calculated or experimental static dependences or simplified relationship like Frugel equations [1] are generally used. However for specific conditions the use of static dependences can result in errors. Fig. 3 shows the change of relative pressure drop in turbine and relative temperature at the turbine output for the constant value of $\frac{G\sqrt{T}}{P}$ with 20% temperature change in 10 seconds. Shown are the data for turbines with various number of stages. These data demonstrate that heat accumulation can substantially influence the dynamics of accident processes. Therefore static dependencies can be used only after the evaluation of dynamic error.

Let us consider the accident process of dynamics caused be the interruption of coolant circulation through the core. Fig. 4 shows variation of temperature in the center of fuel and at the cladding surface following spontaneous loss of coolant circulation and actuation of safety system 1 sec.

thereafter. Due to low coolant heat accumulating capacity the temperature in channels quickly increases. Cladding temperature initially increases much faster because of heat transfer from fuel rod center to cladding and because of practically zero heat removal from cladding. As can be seen from data shown urgent restoration of coolant circulation is nessessary since the fuel rod can be damaged in about 2 minutes.

The study of accident processes following leakage has shown that in case of main loop leakage after turbine (Fig. 5) gradual reduction of coolant flow starting from nominal value occurs. Pressure reduction rate is 2,2 bar/sec. Coolant flow rate in 20 seconds after start of leakage is reduced to 50% of nominal value [2].

In case of rupture of one of six main steam lines at the reactor outlet an abrupt increase of coolant flow rate occurs with subsequent fast reduction in time. Even in this case coolant flow safe in 20 seconds is 50% of nominal value. However in an emergency considered, the pressure drop between upper and lower reactor chambers increases substantially. Such a pressure drop can lead to deformation and destruction of reactor structural elements as well as to compression of fuel elements caused by bending of fuel assembly's casing.

To eliminate the above mentioned phenomena in a steam pipeline at the out-let of the reactor it is advisable to install flow rate limiters. Fig. 5 shows time dependence of coolant flow data through the core with the flow rate limiters. As the data demonstrate the initial coolant flow rate through the core is substantially reduced and is followed by more gradual decrease in time.

Ruptures of the main circuit pipelines at the reactor inlet are the most dangerous, becouse in this case the rate of coolant flow rate through the core drops down rapidly and its reverse circulation appears practically at the same moment. More over large fluctuations of pressure take place in the reactor. In order to prevent reverse circulation of coolant through the reactor it is necessary to increase the number of feeding pipelines. They should be as many as it is necessary so that the break-down of one of them would not lead to the reverse circulation and to inadmissible decrease of the flowrate.

Thus, in case of ruptures of the main circuit pipelines the constant circulation in the circuit maintains for a long period of time, which enables to take measures for reactor shutdown and for connection of emergency cooling systems. The reserve of coolant in the circuit is provided by use of gas - liquid thermodynamic cycle and a large drop between maximal and minimal pressures in the circuit provides driving forces for natural flow of coolant. The rate of pressure increase at gas - liquid cycle with dissociating coolant is approximately 70 at pressure drop up to 160 bars for NPP with fast reactors. These values are considerably lower for other coolants and two-loop circuits.

These features create favourable conditions for natural coolant flow in case of power loss of the main circulating pump and the whole NPP.

Fig. 6 shows the nature of temperature changes for such an accident. Shown here is a change of coolant temperature at the reactor inlet and changes of fuel and cladding temperatures. The temperature of coolant at the reactor inlet decreases because of the pressure decrease at the part of the circuit between the pumps and the reactor and because of sharp decrease of heating in the regenerator die to decrease of flow rate on the heating side. First the temperatures of fuel and cladding decrease sharply due to the increase of flow rate and then they begin to grow.

The analysis of emergency situations with gas-cooled reactors shows that it is necessary to provide for the continuity of coolant circulation in the core for all accident conditions.

In this connection an Emergency Cooling System (ECS) should meet the following main requirements:

- maintaining continuity of coolant circulation whem shifting to cooling with ECS;
- removal of residual heat and cooling down at permissible rates;
- localization of fission products in case of leaking fuel elements.

Fig. 7 shows the diagram of ECS, which meets the above given requirements. During normal operation of a NPP the system is heated and remains in a stand-by regime. In case of emergency situation the fast operating valve 5 is opened by the emergency protection signal and cutt-off valves 2,9 of the main circuit are closed. The coolant accumulated in the regenerator-evaporator 4 flows through the reactor 1 due to a difference of pressures and flows down through the valve 5 into the reservoir in the main circuit with lower pressure, for instance into the condenser. Similarly the maintenance of the continuity of coolant circulation in the reactor is provided at the moment of shifting to ECS cooling. During passive flow of coolant the pump 7 starts up, the condenser 6 is brought into operation and after that the valve 5 is closed down. The cooling system changes into gas-liquid cycle. The coolant circulates in closed circuit (pump-regenerator-condenser-pump). When the temperature of the reactor components comes down to the values, at which thermal shocks are impossible, the coolant is pressed out from the regenerator-evaporator 4 into the circuit (not given on Fig. 7) and cooling is provided by natural circulation in a liquid phase.

Fig. 8 shows possible ECS operation at supercritical pressure and residual heat energy of 0,5-5% Nnom.

The restrictions are: maximal possible temperature, maximal flow rate of coolant after air flushing and minimal superheating of coolant in the regenerator.

At the mode of possible operation the following control criteria have been examined: constant pressure of coolant; constant temperature of coolant at the reactor outlet; constant flow rate of coolant.

Fig. 8 also shows the change of main parameters of cooling system for these control criteria. As is shown by the data, the simplest mode of control criterion is the maintenance of constant flow rate of coolant in the circuit. The process of cooling can be designed in such a way that the rate of temperature change of reactor components will not be higher than permissible values. ECS can be operated for a long period of time in gas-liquid cycle without shifting to liquid coolant.

REFERENCES

1. SHAROVAROV, G.A. Dynamics of nuclear power plants with dissociating coolant. - Minsk.: Nauka i tekhnika, 1980. - 238 pp.

2. SHAROVAROV, G.A., BERNATSKAYA, A.M., ZENICH, T.S., NICHIPOR, V.V. Dynamic characteristics of nuclear power plant equipment and emergency cooling systems. - Proceedings of Byelorussian SSR Academy of Sciences, physics and energy series, 1982, No. 3, pp. 43-47.

3. SHAROVAROV, G.A. The physics of non-steady-state processes in nuclear power plants. - Minsk.: Nauka i tekhnika, 1985. - 203 pp.

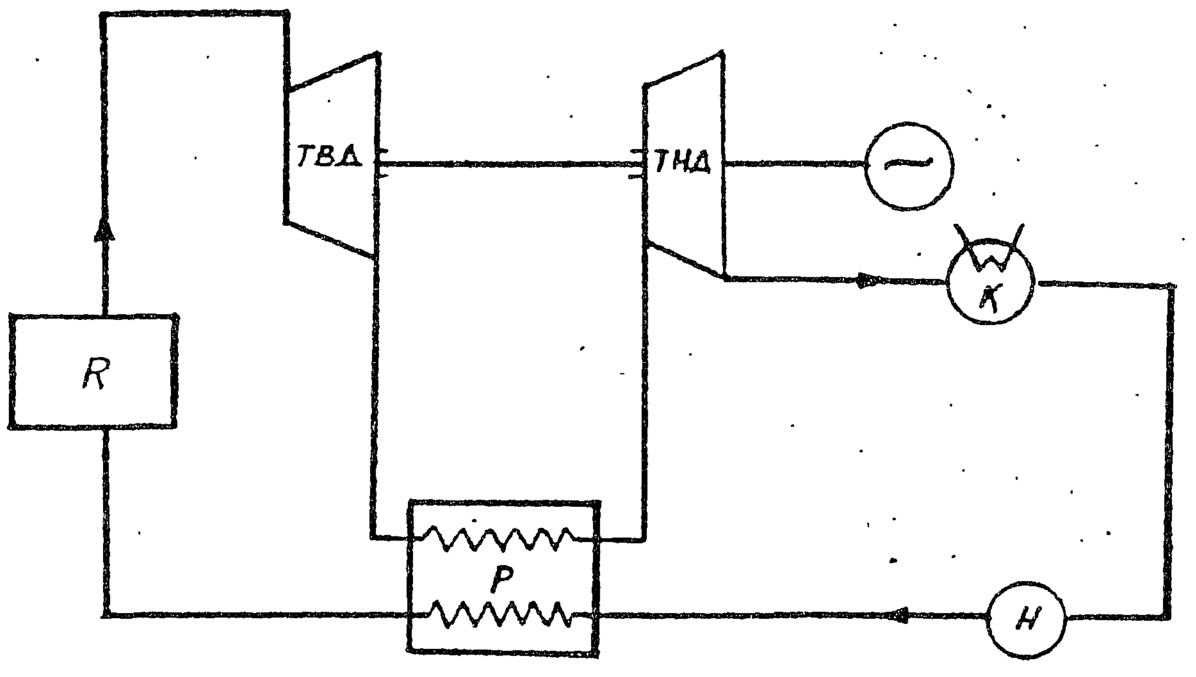


Fig. 1. Technological layout of NPP.

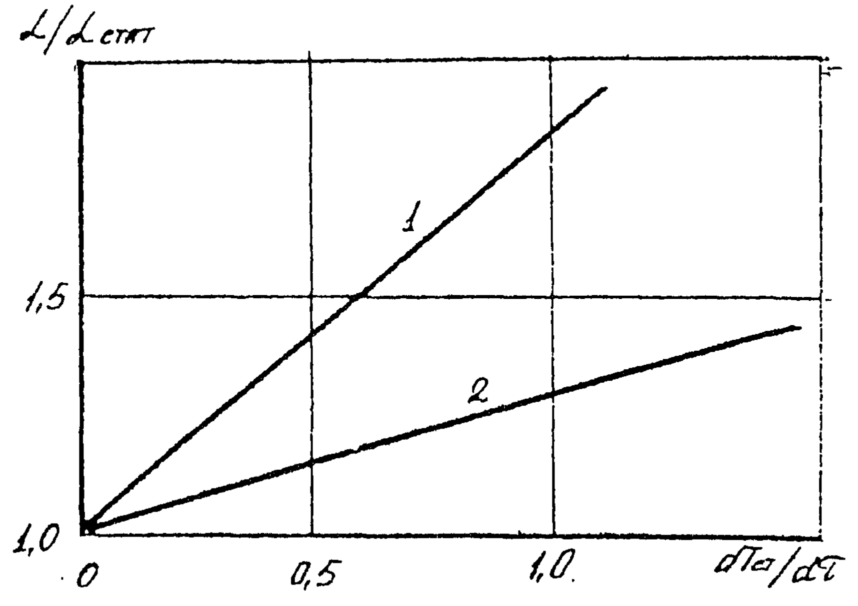


Fig. 2. Non-stationary heat transfer coefficient versus fuel rod cladding temperature change rate: 1 - noble gases, 2 - dissociating coolant.

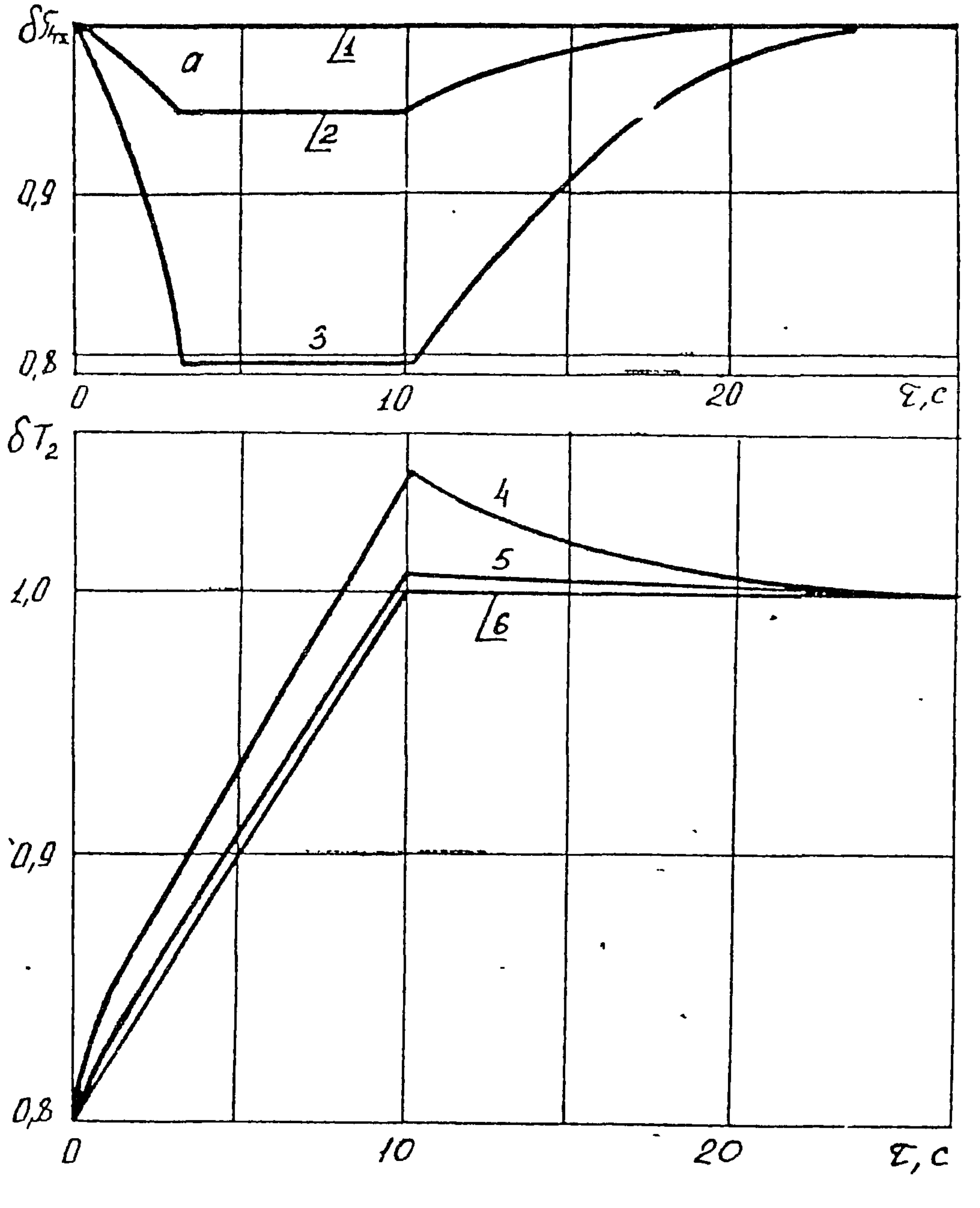


Fig. 3. Relative total pressure reduction rate (a) and relative output temperature versus linear in put temperature change: 1 - z = 6; 2, 5-8; 3, 4 - z = 10; 6 - according to statistical relationships.

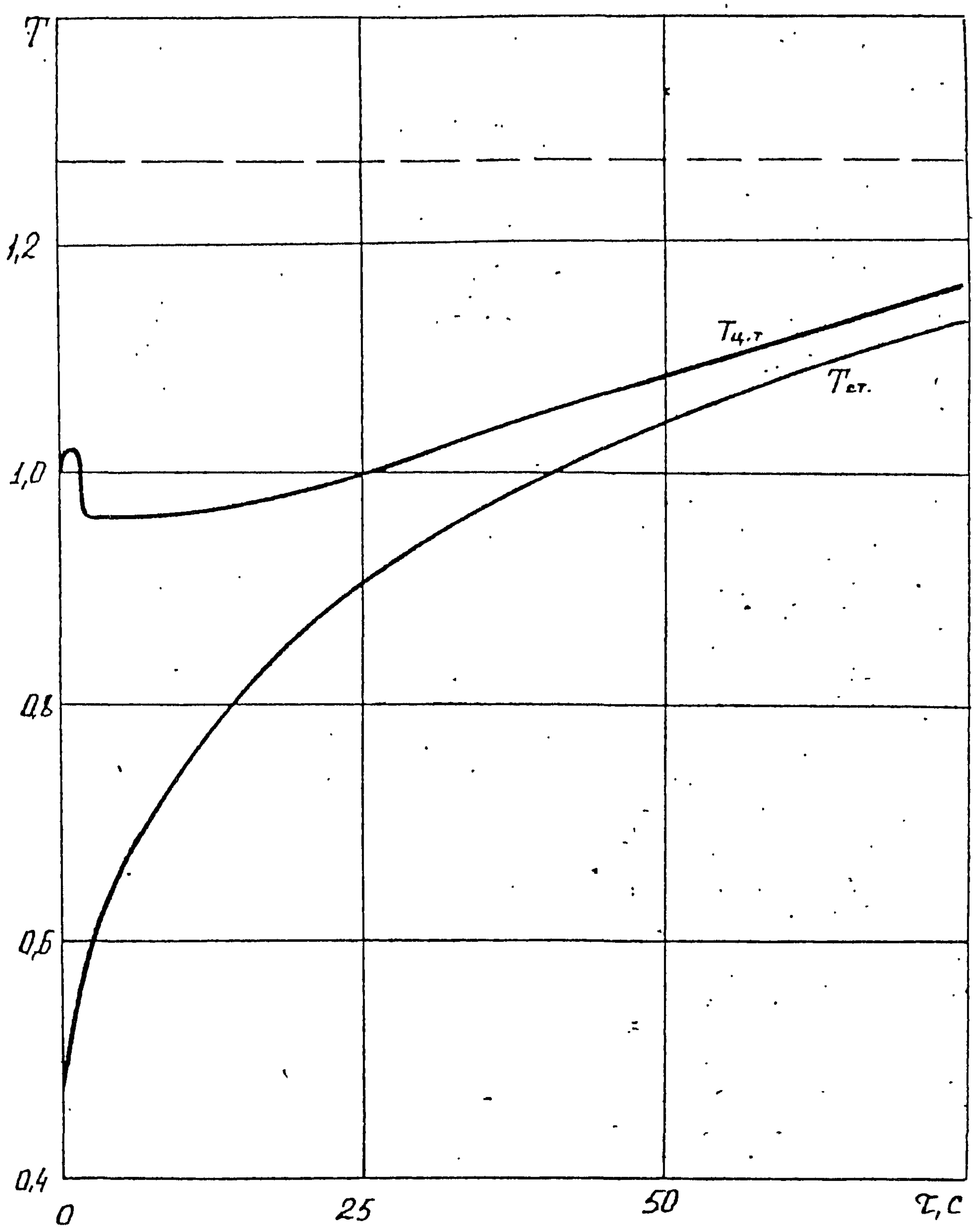


Fig. 4. Variation of temperature in the center of fuel and at the cladding surface following complete loss of coolant circulation.

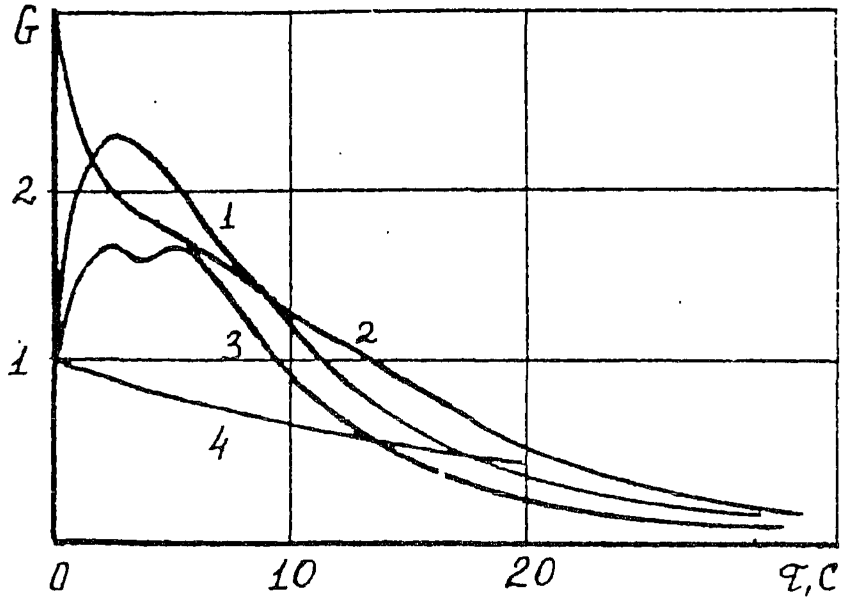


Fig. 5. Time dependence of relative coolant flow rate following reactor outlet pipeline rupture:

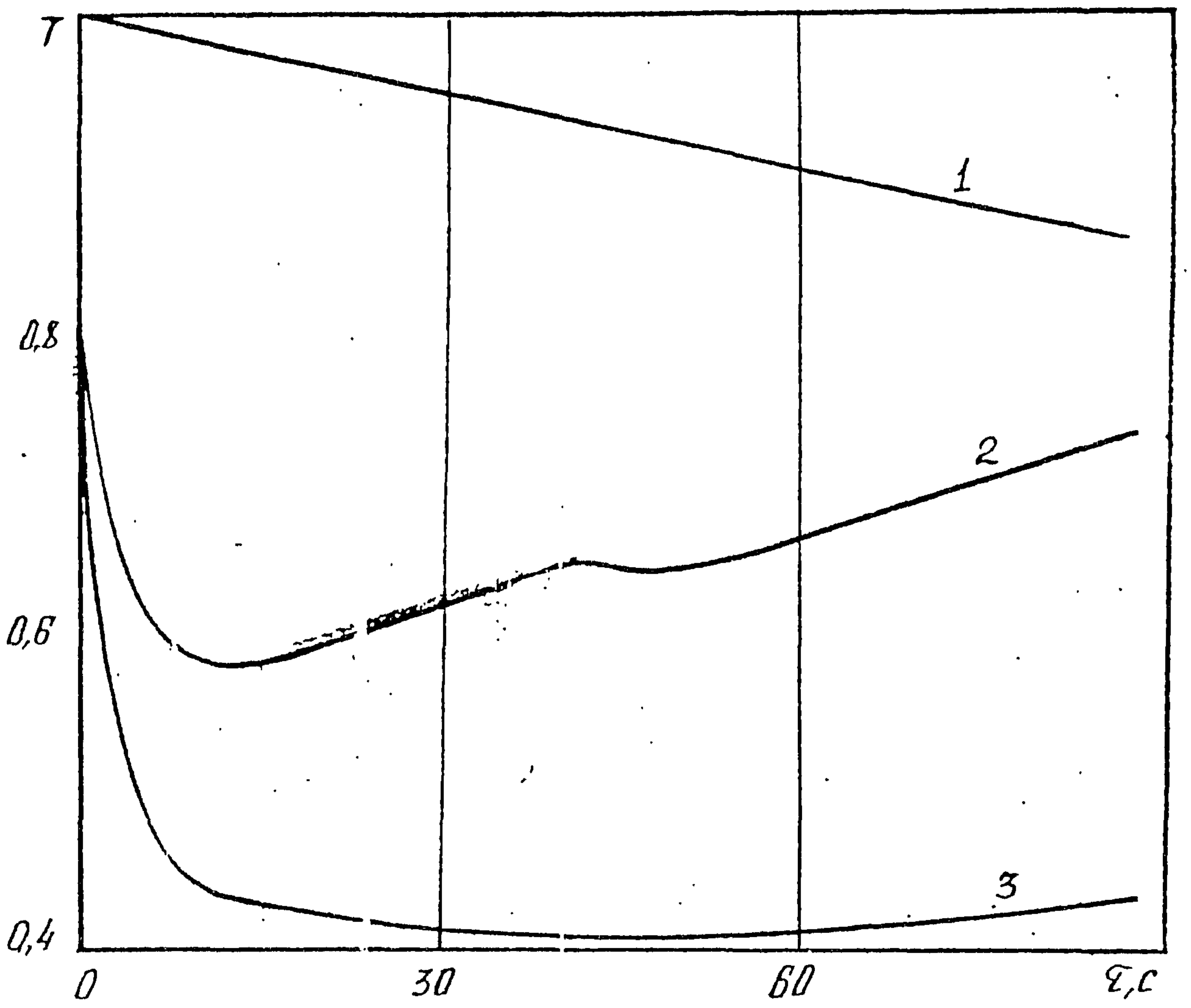


Fig. 6. Changes of temperatures for cladding, fuel and coolant at reactor input.

1 - reactor flow rate without flow rate limiter;

2 - leakage flow rate without flow rate limiter;

3 - reactor flow rate without flow rate limiter;

4 - reactor flow rate with ruptured pipeline after turbines.

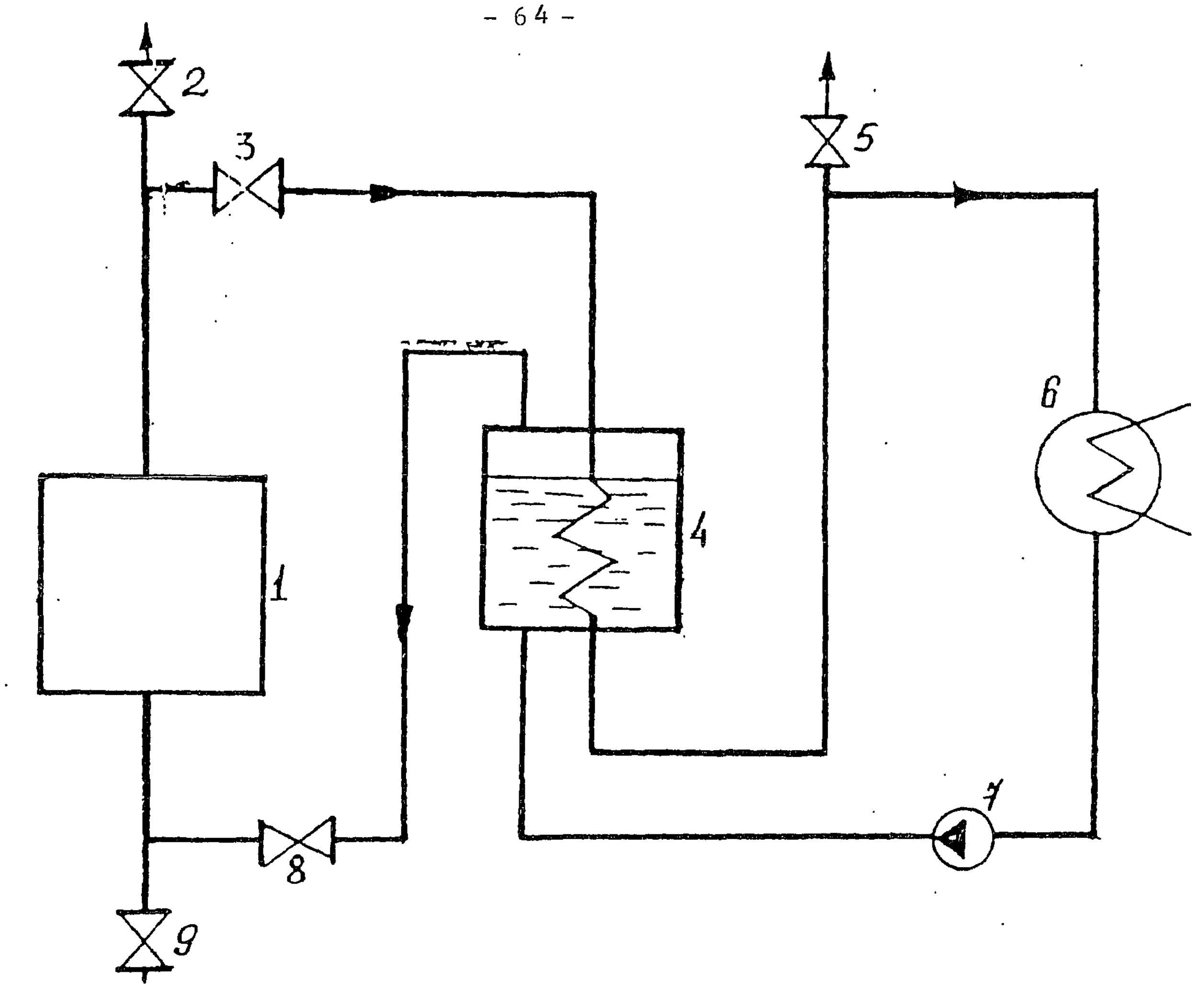


Fig. 7. The block-diagram of the Emergency cooling system: 1 - reactor; 2, 9 - cut-off valves of the main circuit; 3, 8 - cut-off valves of the system; 4 - regenerator-evaporator; 5 - fast operating valve; 6 - condenser; 7 - pump.

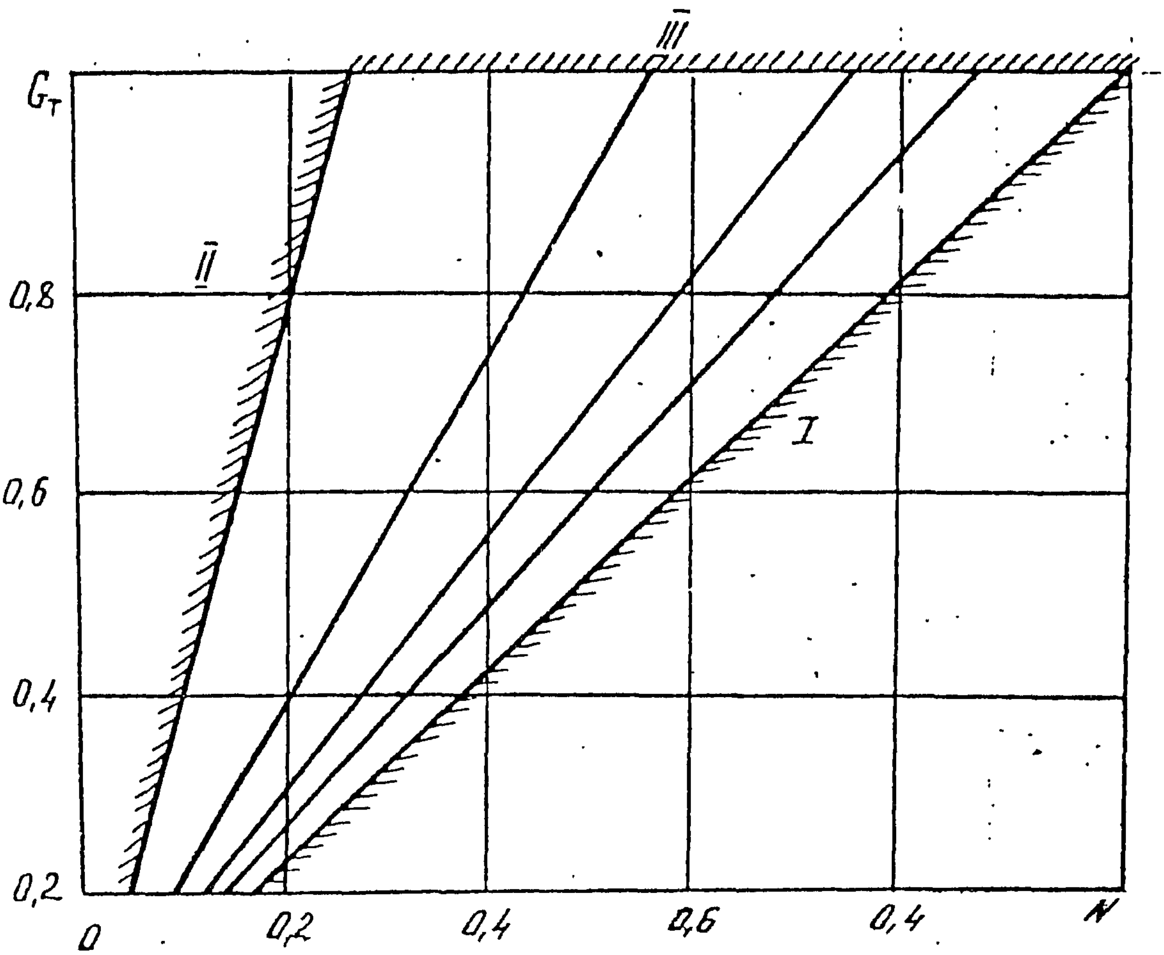


Fig. 8. The mode of possible operation of the Emergency cooling syctem: I - maximal temperature of coolant in the system; II - minimal rate of superheating of coolant; III - constant pressure. I - 3 - constant temperature of coolant, I - = 0,894; 2 - 0,761; 3 - 0,628.

NUCLEAR SAFETY AT SPANISH NUCLEAR POWER PLANTS

Nuclear Safety Council (Spain)

Contents

1. INTRODUCTION

This report describes, in the light of the law governing the creation of the Nuclear Safety Council (15/80) of 22 April and the regulations governing nuclear and radiation-containing facilities (Decree 2869/72) of 21 July, the legal framework and licensing process pertaining to nuclear power plants, together with the mechanisms, responsibilities and functions of the Nuclear Safety Council (CSN) in ensuring the construction and safe operation of the power plants mentioned.

A second aspect dealt with is the technical and legal status of different Spanish power plants, with a brief description of the stage they have reached, their performance during operation and the reassessment processes which they have undergone in applicable cases.

2. LEGAL FRAMEWORK

In Spain nuclear activities are structured from the standpoint of the administrative process and of nuclear safety within a legal framework defined by four basic laws and decrees:

- Law governing nuclear energy (25/64) of 29 April;

- Regulations governing nuclear and radiation-containing facilities: Decree 2869/72 of 21 July (RINR);

- Law on the creation of the Nuclear Safety Council (15/80) of 22 April;

- Statute of the Nuclear Safety Council: Decree 1157/82 of 30 April.

The Nuclear Energy Law made the Ministry of Industry (at present the Ministry of Industry and Energy) responsible for the administrative process for licensing nuclear and radiation-containing facilities. At that time the technical body responsible for evaluation was the Nuclear Energy Board (JEN), which comes under the Ministry of Industry. The law on the creation of the Nuclear Safety Council introduces important changes into the process for licensing nuclear and radiation-containing facilities, among which are the following:

- The CSN is designated as an independent organization of the central administration of the State;

- Safety analysis falls within the sole competence of the CSN and the licensing reports of the latter are mandatory with the force of refusal or conditional acceptance;

- The administrative licensing process for the nuclear and radiation-containing facilities, which is regulated by the RINR, comes under the Ministry of Industry and Energy, which has to send the CSN a request for the evaluation of safety-related matters.

3. NUCLEAR POWER PLANT LICENSING PROCESS

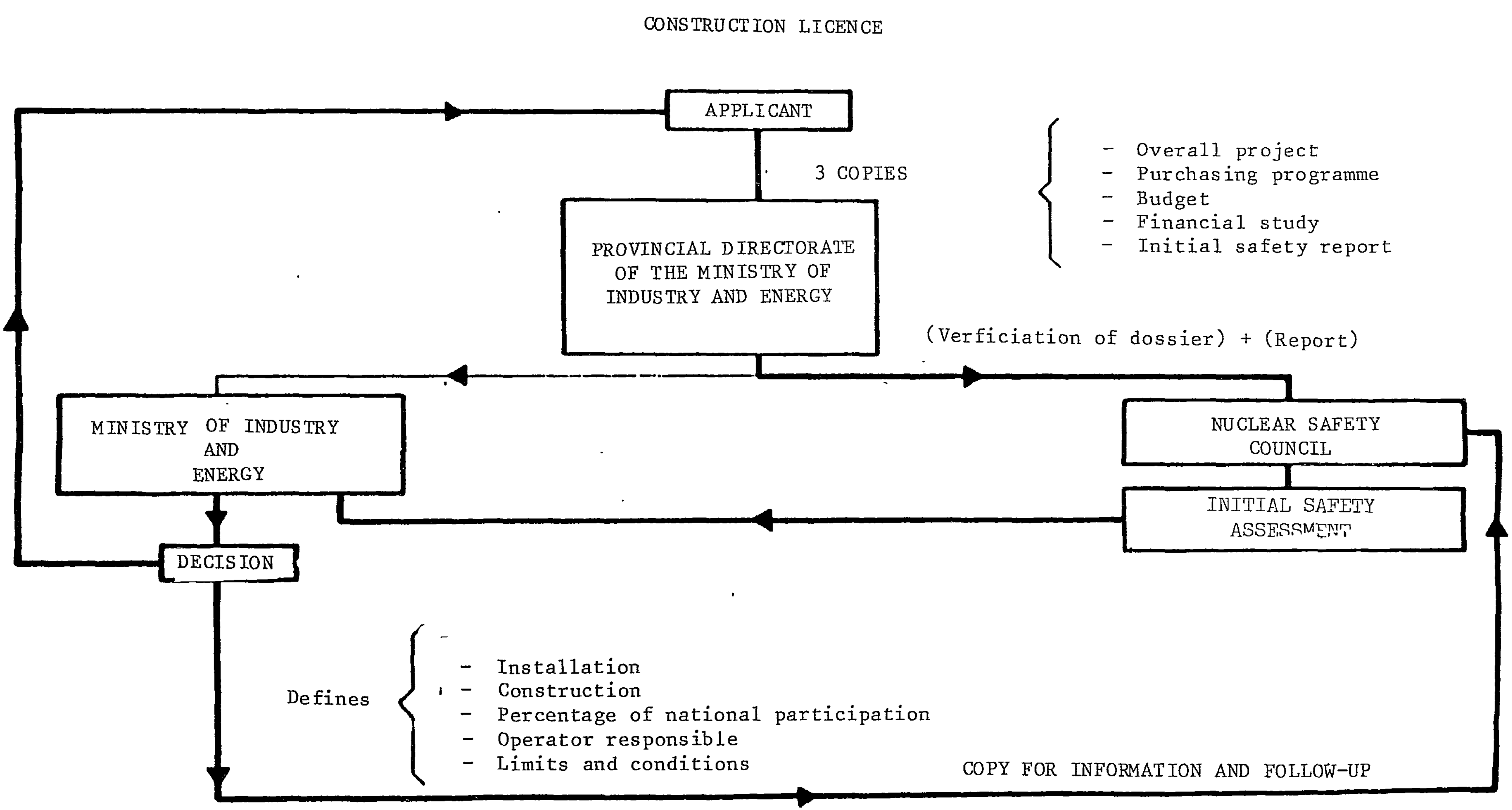
The administrative licensing process for nuclear and radiation-containing facilities is defined in the Regulations governing nuclear and radiation-containing facilities of 1972, which are now at a very advanced stage of revision. The new version, which is due to be published this year, will take into account the directives of the European Communities and the recent Spanish legislation on adoption and standardization, as well as covering regulation of the activities of the firms offering services and technical assistance that the applicant may resort to in support of the application for each licence.

According to the RINR, the following are the licences or permits required by a nuclear power plant:

- Preliminary licence;
- Construction;
- Pre-operational inspection;
- Interim storage of nuclear materials;
- Provisional operation or start-up;
- Alteration (where applicable);
- Final operation;
- Shut-down.

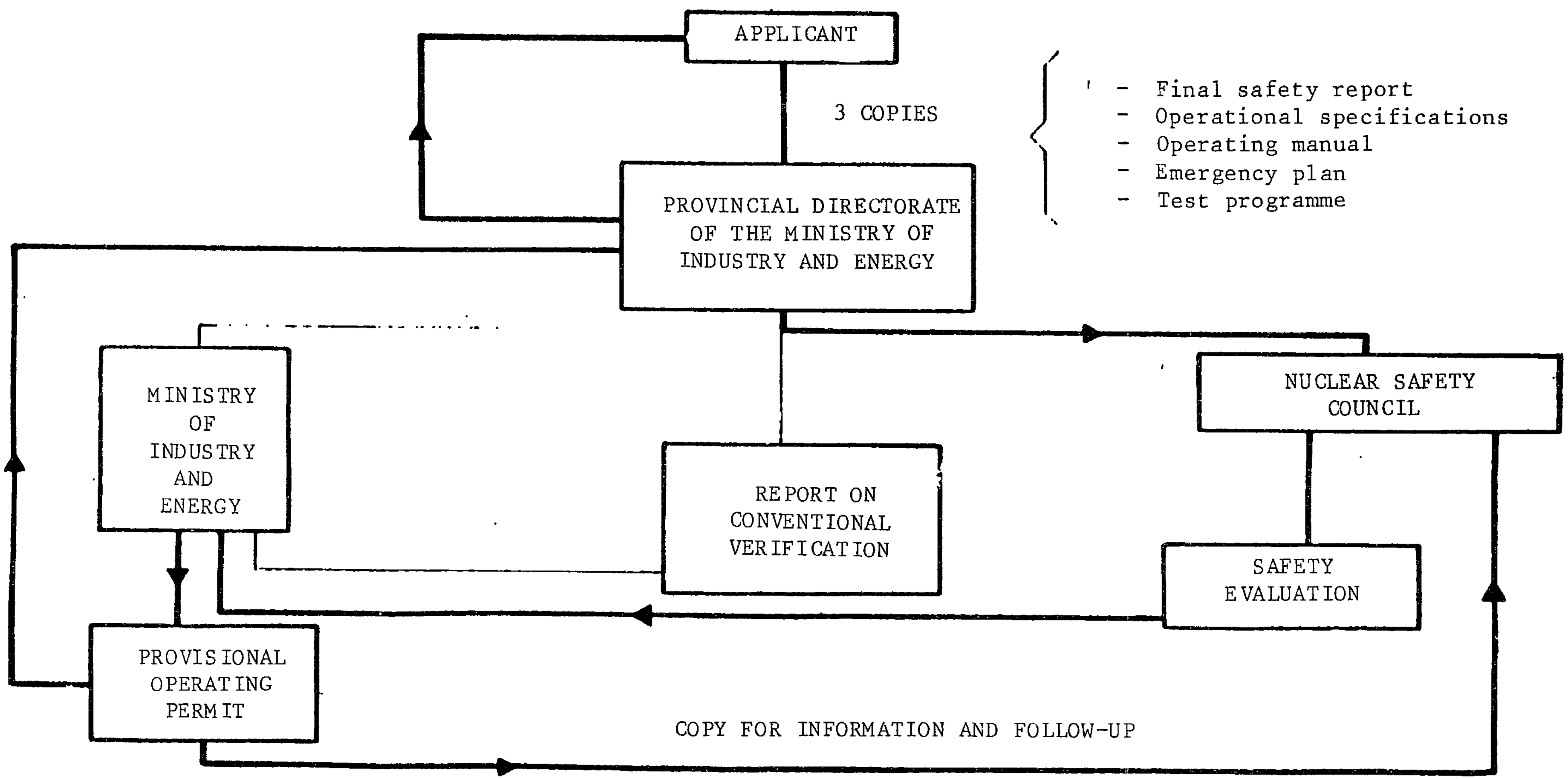
Furthermore, likewise in accordance with the RINR, there must be a licence for the manufacture of different safety-related components of the power plant, and certain plant personnel (those in charge of shifts and control room operators as well as the head of the radiation protection service) must possess the appropriate licence issued by JEN (now the CSN).

The process to be followed for obtaining the different licences can be summed up in the form of the following flow charts:

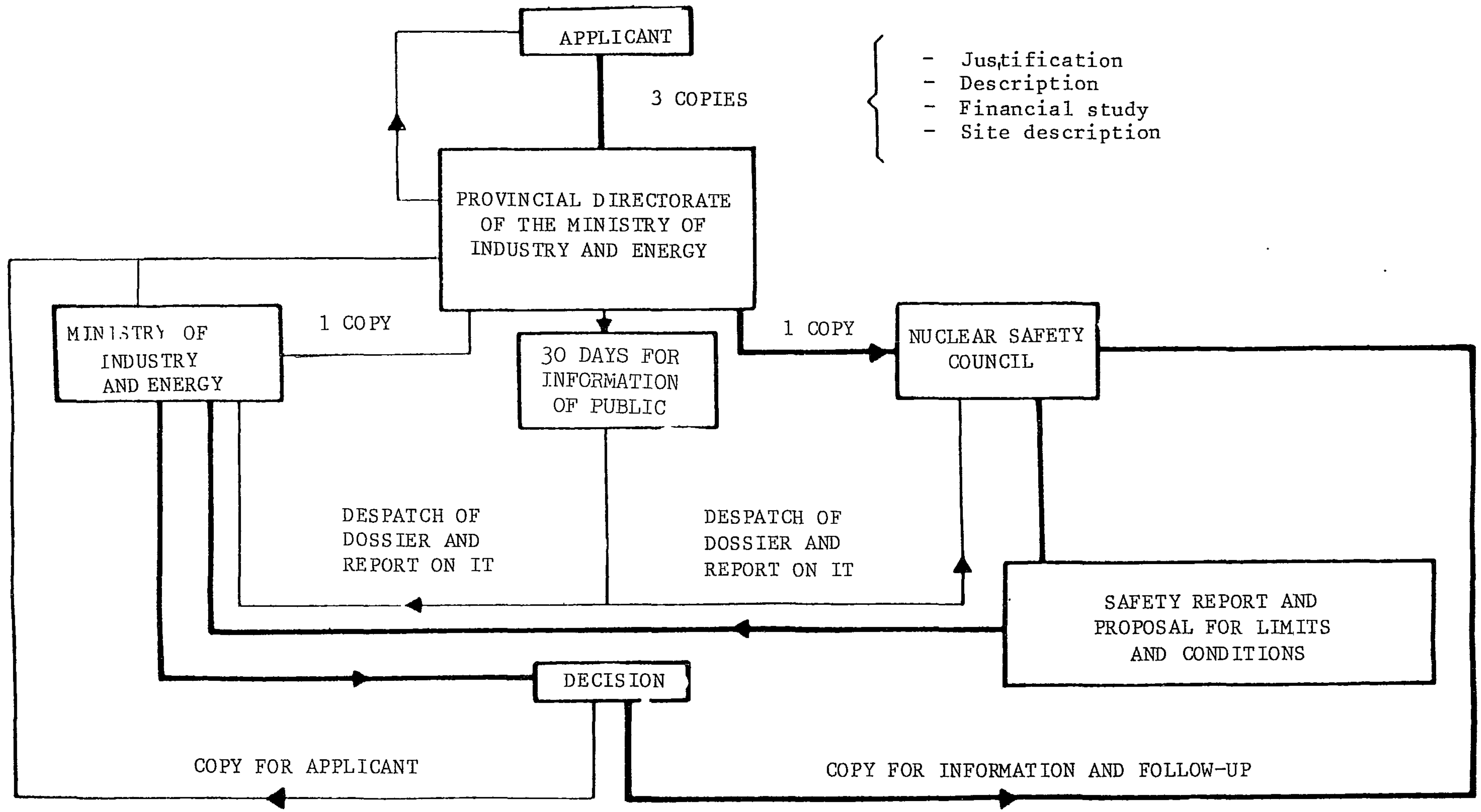


CSN/DEL 040187

PROVISIONAL OPERATING PERMIT (NUCLEAR AND RADIATION-CONTAINING FACILITIES - CATEGORY I)



PRELIMINARY LICENCE (NUCLEAR AND RADIATION-CONTAINING FACILITIES - CATEGORY I)



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4. STATUS AND CHARACTERISTICS OF THE NUCLEAR POWER PLANTS

The set of nuclear power plants presently at the operational or construction stage represents three generations that are differentiated within the nuclear programme:

First generation: Power plants planned in the 1960s, the construction of which was complete by the end of that decade or the beginning of the 1970s. This generation includes the "José Cabrera" plant, which began operation in 1969; the "Santa Maria de Garoña", which started up in 1971, and the "Vandellós I" plant which commenced operation in 1972.

Second generation: Power plants planned at the beginning of the 1970s, the construction of which began during the same period and which were scheduled to be put into operation by the end of the decade. Delays in the construction process meant that the first of these did not go into commercial operation until 1981. This generation includes the "Almaraz I and II" nuclear power plants, which were put into operation in 1981 and 1983, respectively, the "Asco I and II" plants, which were started up in 1982 and 1985, and the "Cofrentes", which began operation in 1984. Construction of the "Lemoniz I and II" units, which also belong to this generation, has been stopped.

Third generation: Nuclear power plants whose construction was licensed after approval of the National Energy Plan in July 1979; they were planned at the end of the 1970s and construction was begun in 1979. This generation includes the "Trillo I" and "Vandellós II" plants, which are scheduled to begin operation early in 1988, and the "Valdecaballeros I and II" and the "Trillo II" plants, on which construction has been stopped.

The technical characteristics of these power plants are shown in the attached table.

TABLE OF TECHNICAL CHARACTERISTICS

NAME AND TYPE OF PLANT	JOSE CABRERA (PWR)	S.M.GAROÑA (BWR)	VANDELLOS I (GCR)	ALMARAZ I - II (PWR)	ASCO I - II (PWR)	COFRENTES (BWR)	LEMONIZ I - II (PWR)	VALDECA I - II (BWR)	TRILLO I - II (PWR)	VANDELLOS II (PWR)
POWER										
. Thermal (MW(t))	510	1 380	1 670	2 696	2 696	2 894	2 696	2 894	3 010	2 785
. Gross electrical (MW(e))	160	460	500	930	930		930		1 041	982
. Net electrical (MW(e))	153	440	480			974		974		930
REACTOR										
. Type of fuel	UO_2	UO_2	U natural	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
. Coolant	Pressurized water	Water-steam	CO_2	Pressurized water	Pressurized water	Water-steam	Pressurized water	Water-steam	Pressurized water	Pressurized water
. Moderator	Water	Water	Graphite	Water	Water	Water	Water	- Water	Water	Water
. Inside diameter of vessel (cm)	289	477.5	1 900	398.8	398.8	544	398.8	544	487.8	398.8
. Overall inside height of vessel (cm)	874	1 844.7	3 620	1 299.2	1 298.0	2 090	1 298	2 090	1 103.9	1 217.7
. Design presssure of vessel (kg/cm^2)	175	87.9	30.4	174	173.8	87.88	173.8	87.88	172.5	144.7
. Coolant temperature at vessel input (°C)	281	183	220	291	291	215	291	215	292	290
. Total flow rate through core (kg/h)	17.03×10^6	21.77×10^6	32.7×10^6	47.6×10^6	47.6×10^6	38.33×10^6	47.6×10^6	38.33×10^6	53.78×10^6	48.76×10^6
CORE AND FUEL										
. Core diameter (cm)	185.8	368.3	1 370	304	304	429.5	304	429.5	345	304
. Fuel material	UO_2 pellets	UO_2	Natural U tube	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2
. Number of fuel elements	69	400	46 080 (max) 1/	157	157	624	157	624	177	157
. Arrangement of fuel rods	14 x 14	8x8		17 x 17	17 x 17	8 x 8	17 x 17	8 x 8	16 x 16	17 x 17
. Active fuel length (cm)	243	368.9	900 2/	362.1	366	423	366	375.9	340	362
. Cladding material	Zircaloy-4	Zircaloy-2	Mg-Zr alloy	Zircaloy-4	Zircaloy-4	Zircaloy-2	Zircaloy-4	Zircaloy-2	Zircaloy-4	Zircaloy-4
. Cladding thickness (cm)	0.0617	0.081	0.18	0.056	0.057	0.081	0.057	0.086	0.0725	0.05715
. Channel material (BWR)	-	Zircaloy-4 -	-	-	-	Zircaloy-4	-	Zircaloy-4	-	-
CORE DESIGN CONDITIONS										
. Core power density (kW/l)	78.7	40.6	1.04	101.2	101.2	52.4	101.2	52.4	93	109.2
. Heat transfer surface (m^2)	951.2	3 600	64 212	4 515.1	4 515		4 515	5 515	4 796	4 515.87
. Mean heat flow (cal/h-m^2)	42.7×10^3	744.9×10^3	-	500.6×10^6	500.6×10^6	419.4×10^6	500.6×10^6	432.7×10^6	527.9×10^6	514.6×10^6
. Maximum heat flow (cal/h-m^2)	79.1×10^3	$1\ 084.6 \times 10^3$	888×10^6	$1\ 161.4 \times 10^6$	$1\ 161.4 \times 10^6$	980.9×10^6	500.6×10^6	960.2×10^6	$1\ 409.2 \times 10^6$	$1\ 194.3 \times 10^6$
. Critical power limit ratio (safety limit for boiling)	1.93	1.06	-	1.30	1.30	1.06	1.30	1.06	1.3	1.3
. Mean void factor for coolant inside fuel element during the cycle (BSR)	-	40%	-	-	-		-	43%	-	-
. Maximum rate of linear heat generation (W/cm)	535	440		400.3	629.3	439.6	629.3	439.6	582.14	413.38
CONTROL SYSTEM										
. Number of control rods	272	97	72 grey + 63 black	48 bundles	48 bundles	145	48 bundles	145	52 bundles	48 bundles
. Shape of control rods	Spider	Cross-shaped	Tubular-double tube	Spider	Spider	Cross-shaped	Spider	Cross-shaped	Spider	Spider
. Control rod grid	14 x 14	365.8	Centred hexagonal	24 rods/bundle	24 bars/bundle	19 tubes in each wing	24 rods/bundle	19 tubes in each wing	20 rods	24 rods/bundle
. Control rod material	Ag80-In15-Cd5	C_3B_4	C_3B_4	Ag80-In15-Cd5	Ag80-In15-Cd5	C_3B_4	Ag80-In15-Cd5	C_3B_4	Ag80-In15-Cd5	Ag80-In15Cd5 & C_3B_4
. Type of actuator	Magnetic	Hydraulic	Electric	Magnetic	Magnetic	Hydraulic	Magnetic	Hydraulic	Magnetic	Magnetic
. Reactor power monitoring system	Burnable poison	Variation in recirculation rate and control rods -	Control rods and variation in coolant flow	Control rods; burnable poison rods; boric acid	Control rods; burnable poison rods; boric acid	Control rods; recirculation flow; burnable poison	Control rods; poison rods; boric acid	Control rods; recirculation flow rate; burnable poision	Control rods	Control rods and boric acid
PRIMARY COOLANT FLOW CIRCUITS										
. Type	Closed loop	Recirculation and jet pumps	Turboblower with heat exchanger	Closed loop	Closed loop	Recirculation and jet pumps	Closed loop	Recirculation and jet pumps	Closed loop	Closed loop
. Position of the circuits	In containment	Dry well	Inside containment	At 120° in containment	At 120° in containment	Dry well	At 120° in containment vessel	Dry well	At 120° in containment	At 120° in containment
. Number of circuits	1	2	1	3	3	2	3	2	3	3
. Piping diameter (cm)	69.8	61	-	69.85 (int)	69.8 (int)	50.8	69.8 (inside)	50.8	75	
. Pump capacity (m^3/min per unit)	295.2	121.1	2 370 kg/s	356	356	124.6	356	124.8	428	86.36
. Number of jet pumps (BWR)	-	20	-	-	-	20	-	20	-	-
. Position of the jet pumps (BWR)	-	Inside containment	-	-	-	Inside containment	-	Inside containment	-	-
. Jet pump pressure (BWR) (kg/cm^2)	-	70.3		-	-		-			

TABLE OF TECHNICAL CHARACTERISTICS (cont'd)

NAME AND TYPE OF PLANT	JOSE CABRERA (PWR)	S.M.GAROÑA (BWR)	VANDELLOS I (GCR)	ALMARAZ I - II (PWR)	ASCO I - II (PWR)	COFRENTES (BWR)	LEMONIZ I - II (PWR)	VALDECA I - II (BWR)	TRILLO I - II (PWR)	VANDELLOS II (PWR)
STEAM GENERATORS			EXCHANGER							
. Type	Vertical U tubes		Single tube in vertical panels	Vertical U tubes	Vertical U tubes		Vertical U tubes		Vertical U tubes	Vertical U tubes
. Number	1		1	3	3		3		3	3
. Unit steam flow rate (kg/sec)	268		604.4	500	500		500		550	512.9
. Heat transfer surface (m^2)	2 307		132 833	44 593	44 592		44 592		5 386	4 487
. Water input temperature (^oC) (primary) ..	316		87.7 3/	343	343		343		325.7	221.89
. Steam output temperature (^oC)	259		388.8	284.2	284		284	-	284.5	282.22
POWER PLANT INSTRUMENTATION SYSTEM										
. Position of neutron detection system	Outside	In-core	Detectors behind leaktight covering	Outside	Outside	In-core	Outside	In-core	Outside	Outside
. Nuclear instrumentation ranges (above rated power)										
.. Starting range	$4x10^{-2}$-$4x10^4$*	Source to 0.01%	GkW - 12 MW	10^{-1} to 10^5*	10^{-1} to 10^5*	Criticality outage	10^{-1} to 10^5*	Criticality outage	5.10^{-3} to 5.10^4	10^{-1} to 10^5*
.. Intermediate range	250-$2.5x10^{10}$	0.0001% - 10%	12 - 300 MW	10^2 to 10^{10}*	10^2 to 10^{10}*	Crit.-10 % power	10^2 to 10^{10}*	Crit.-10% power	10^3 to 10^{10}	$5x10^2$ to 10^{11}*
.. Power range	2 500-$2.5x10^{10}$	1 - 125%	100 - 1800 MW	10^7 to 10^{10}*	10^7 to 10^{10}*	1-125% power	10^7 to 10^{10}*	1-125% power	5.10^7 to 10^{10}	$5x10^7$ to 10^{11}*
REACTOR PROTECTION SYSTEM						-				
. Number of channels (signal groups)	4	4 (2 x log)	2	2, 3 or 4	2, 3 or 4	4 (2 logic)	2, 3 or 4	4 (2 logic)	2, 3, or 4	2, 3 or 4
. Number of channels needed in case of emergency	2	2 (1 x log)	2 out of 3	1 out of 2, 2 out of 3 or 2 out of 4	1 out of 2, 2 out of 3 or 2 out of 4	2 (1 logic)	1 out of 2, 2 out of 3 or 2 out of 4	2 (1 logic)	1 out of 2 or 2 out of 4	1 out of 2, 2 out of 3 or 2 out of 4
. Number of channels for other protection functions	2 - 6	2	2 out of 3	1 out of 2, 2 out of 3 or 2 out of 4	1 out of 2, 2 out of 3 or 2 out of 4	4	1 out of 2, 2 out of 3 or 2 out of 4	4	4	2, 3, or 4
. Number of sensors for each variable monitored in each channel	2 - 6	2	2 out of 3	1 per channel	1 per channel	1, 2, or 4	1 per channel	1, 2, or 4	1 per channel	1 per channel
. Method of preventing accidental removal of control rods	Locking	Automatic locking (RBM)	Locking	Automatic and manual locking	Automatic and manual locking	RPCS below 20% power + RBM	Automatic and manual locking	RPCS below 20% power + RBH	Automatic locking	Automatic locking
CONTAINMENT										
Type (primary, secondary)	Primary	Primary and secondary	Vessel	Primary	Primary	Primary + secondary	Primary	Primary + secondary	Primary	Primary
. Primary containment	Building	Dry and wet well	Vessel	Building	Building	Dry and wet wells and containment	Building	Dry and wet wells and containment	Building	Building
.. Dry well design pressure (BWR) psig .	-	62	-	-	-	30	-	30	-	-
.. Wet well design pressure (BWR) psig .	-	62	-	-	-	15	-	15	-	-
.. Containment building pressure (PWR) psig	32	-		50	54	-	55	-	76.8	-
.. Design leakage rate in primary containment	0.1%/d	0.5%/d at 62 psig		0.10%/d U-1 0.15%/d U-2	0.10%/d	0.5%/d	0.10%/d	0.4%/d	0.25%/d	
.. Thickness of primary containment steel (inches)	0.25	5/8 - 2	10 (leaktight) covering)	0.393	6.5 mm (0.26 inch)	1.5 inch	0.302 inch	1.5	1.18	0.256
.. Thickness of primary containment concrete (cm)	91.5	-	Walls 425 Bottom and top >500	Cylinder 140 Dome 75 Slab 350	Cylinder 150 Dome 100 Base 275	227 (only in lower part)	Cylinder 114 Dome 101 Slab 533	-	-	Cylinder 115 Dome 95 Slab 300
(1) Natural uranium cylinder with graphite content										
(2) Length of 15 tapered elements										
(3) Water for steam production (secondary) ..										
(*) $\frac{neutrons}{cm^2.sec}$										

5. SAFETY OF THE NUCLEAR POWER PLANTS

The uninterrupted development of safety requirements is the result of evaluation and constant appraisal of the criteria and procedures employed to ensure the safe operation of the nuclear power plants. A determination of what is needed to improve the safety of a facility is made on the basis of results from:

- Nuclear safety research;
- Operational experience at both domestic and foreign facilities;
- Evaluation of facilities in terms of the current standards applied by advanced countries;
- Appraisal of facilities based on the use of new techniques offering different standpoints with regard to facility safety.

It is important to point out that the CSN is pursuing activities in the four areas mentioned above by participating in international research programmes* and in the promotion of research at national level, collaboration with bodies analysing events occurring at power plants abroad and analysis of events at Spanish facilities, evaluation of facilities with several years of operation in terms of the standards currently applied in advanced countries, and the employment, for example of probabilistic safety analysis.

In this respect we should point out that some of the more significant changes which have occurred in Spanish nuclear power plants have resulted from activities involving evaluation of the facilities in terms of the present nuclear safety standards (general alterations at the José Cabrera plant), our own operating experience (replacement of the recirculation system at the Santa María de Garoña), and the use of new techniques, such as probabilistic safety analysis.

There can be no doubt that an exceptionally important source for learning and improving matters is domestic and foreign operational experience, derived not only from events that have greatly influenced public opinion, but also from those that have had little or no impact but which are indicative of safety problems that were not considered before.

The operational specifications for nuclear power plants require the CSN to be informed of any incidents that occur, so that not only the operator can make an analysis of the implications of them, but also so that the CSN can review those that are most significant.

The CSN is a member of such international organizations as the Incident Reporting System (IRS) of the OECD and the IAEA. In addition, it has signed co-operation agreements and maintains close contact with bodies in other countries that have the same duties and responsibilities.

The operators of Spanish nuclear power plants participate in the work of organizations such as the American Institute of Nuclear Power Operations (INPO), which permits the exchange of specific experience between power plants in different

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* Some of the international programmes in which the CSN is taking part are:

PISC - Programme for Inspection of Steel Components
IPIRG - International Piping Integrity Research Group
LACE - Light Water Reactor Aerosol Containment Experiments
LOFT - Loss of Fluid Test

countries. Experience is also exchanged by means of such organizations as the International Union of Producers and Distributors of Electrical Energy (UNIPEDE) and in the form of direct contacts between enterprises.

In addition to the analysis of operational experience on a permanent basis there is need for more detailed examination of serious incidents since the number and importance of the lessons to be learned in such cases is considerable.

This type of analysis has already been made in the past, a clear example being the Three Mile Island-2 accident in 1979. It is important to point out, however, that although many improvements have been made in the light of the lessons learned from that accident, there are still topics on which no final conclusion has been reached, given the complexity of the required analyses and the importance of ensuring that the greatest effort is deployed in areas where the benefits for safety would be greatest.

A number of lessons have also been learned from the Chernobyl accident in the areas of nuclear safety, radiation protection and emergency planning, in which there is need to delve deeper in order to ascertain whether any changes need to be made in any of those three fields.

Within this context the first-generation power plants - José Cabrera and Santa Maria de Garoña - have begun and are already about to complete a process of reappraisal and modification that has taken into account:

- Spanish operational experience and the specific features of the power plant;

- The topics analysed in the Systematic Evaluation Programme (SEP) put into effect by the first-generation power plants of the United States;

- The points emerging from the TMI accident.

This programme has led to important changes such as the improvements to the ECCS and internal power system of the José Cabrera nuclear plant, or the changes in the containment, fire-fighting and power at the Santa Maria de Garoña plant.

The Vandellós I started a programme of re-evaluation and improvement in the spring of 1986, due to last until the end of 1989, which will take into account both Spanish operating experience and that of the reference power plant, in this case the French plant at St. Laurent des Eaux, as well as the relevant topics of the SEP programmes and those based on the TMI accident.

In the case of the second-generation power plants that are now in operation, they will all be obliged to send the CSN, every six months, a status report showing to what extent the requirements of the regulatory body of the country originating the project (in the given case the USNRC) for power plants of similar design have been met. Wherever necessary the appropriate changes have been made or are in process of being made. This has been the case with the alterations deriving from the TMI accident, the improvements to the ATWS mitigation system, or the changes necessary to avoid H_2 explosions in the containment.

The CSN follows the operation of the power plants by means of the information submitted by the operators in accordance with what is stipulated in the operational specifications, as well as through specialized inspections related to set matters. The inspections are stepped up during the trial period of the nuclear test pre-operational phase or during outages for reloading or maintenance. In addition, there is an on-site resident inspector at all the second-generation plants and at some of the first-generation plants. It is the CSN's intention that during the present year 1987 all power plants in operation should have a nuclear safety and radiation protection officer on the spot at all times to check that the limits and conditions set by the CSN are being met.

The attached tables summarize the most important operational data for the Spanish power plants during 1986, together with the development of the Nuclear Safety Council in terms of its resources and activities.

NUCLEAR POWER PLANT OPERATING DATA 1986

	J. Cabrera	Garoña	Vandellós I	Almaraz I	Almaraz 2	Ascó 1	Ascó 2*	Cofrentes
Mean load factor (%)	79.56	88.53	79.56	69.63	74.87	68.39	69.23	74.19
Shut down for refuelling and maintenance (days)	36	9 (in 1986) + 6 months (in 1985)	20	14 (in 1986) + 51 (in 1985)	66	100	-	46
Mean operating factor (%)	86.72	93.3	91.35	74.2	82.1	71.17	75.83	78.58
Number of unscheduled reactor outages	2	6	9	3	9	2	14	4

* First year of operation (nuclear test period).

DEVELOPMENT OF RESOURCES AND ACTIVITIES OF NUCLEAR SAFETY COUNCIL (CSN)

	1981	1982	1983	1984	1985	1986	1987
Budget 10^6 pesetas	72	425	720	882	1 136	1 468	2 015
Inspection of plants under construction	46	68	87	96	38	90	
Inspection of plants in operation	54	20	101	129	162	101	
Installed capacity (MW(e))	2 051	2 051	3 911	4 485	5 815	5 815	
Nuclear power produced (Gwh)	9 586	8 871	10 661	23 086	28 045	37 460	
Total power produced (Gwh)	111 232	114 569	117 196	120 042	127 216	128 560	
Percentage of nuclear power in total output (%)	8.6	7.6	9.0	19.2	22.0	29.13	
Technical personnel working for the CSN (**)		46 + 34	80 + 20	98 + 29	118 + 30	117 + 23	178*

* Estimated for 1987.

** Officials + employees engaged.

NUCLEAR SAFETY IN AUSTRIA

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Nuclear safety in Austria? In a country which renounces nuclear energy? We feel, however, that nuclear safety is an important issue for each country which wants to protect the health of its citizens - not only for those who have nuclear power plants in operation.

In the following I would like to show an example how a small country with very limited resources can develop its own competent expertise in this field.

As a neutral country situated geographically between East and West, Austria feels a special obligation for promoting international collaboration. For a small country this collaboration is also indispensable. Collaboration means not only taking from the international community - it means also giving. Some countries sometimes tend to overlook the latter. Support might be a good thing but collaboration is much better and much more efficient. It will function properly, however, only when everybody is willing to participate actively.

The IAEA, of course, is an important focus of international cooperation and we feel that the tremendous amount of work done there has been most beneficial to nuclear safety.

For us also the Committee for the Safety of Nuclear Installations (CSNI) of the OECD and bilateral agreements are important, as will be shown later.

The goal of health protection needs general nuclear safety experience with a special emphasis on dosimetry and reactor safety. So in Austria the Austrian Research Center Seibersdorf (ÖFZS) together with a number of other institutions is active in the following fields of nuclear safety.

Dosimetry

Within the IAEA/WHO Network of Secondary Standard Dosimetry Laboratories (SSDLs) dosimetry systems are developped and manufactured.
For the 14000 radiation workers in Austria a Personnel Monitoring Service using automated TLD systems is provided.
The control of food samples etc. for radioactivity included more than 70000 samples after the Chernobyl accident.

Reactor Safety

To provide expert opinion - especially for the government but also for the information of the public - in the safety of nuclear power plants of different designs and of reprocessing plants work in reactor safety research is necessary.

This is done in the frame of the IAEA and the Committee for the Safety of Nuclear Installations (CSNI) of the OECD together with a number of bilateral cooperations with countries such as the USA, Federal Republic of Germany, Hungary, Israel, and the People's Republic of China.

So Austria is a member of the IAEA Technical Committee on Thermal Reactor Safety Research and collaborates in the IAEA guidebooks and training courses.

It contributes to the Network of Analytical Laboratories of the IAEA Department of Safeguards and provided the first impulse for the IAEA Regional Project for Europe and the Middle East on Computer-Aided Safety Analysis.

It played an important role in the first IAEA Standard Problem Exercise both in two-phase instrumentation and in the exercise itself. This provided a possibility to become familiar with the safety of WWER-440 type reactors, some of which stand near the Austrian border.

We have, of course, experience with the safety of boiling water reactors because of the licensing work we did for the Austrian nuclear power plant GKT, whose pitiable fate is well known.

Last but not least we are actively involved in the OECD-LOFT experimental program in Idaho Falls, USA, which made us familiar with the safety of pressurized water reactors.

I wanted to show how Austria can serve as an example how a small country can become competent in nuclear safety and how it can try to become a respected member of the international community. I also wanted to show that Austria can be a partner both within the frame of the IAEA and in bilateral agreements. While Austria is certainly not playing the first violin in the international nuclear safety orchestra it does its best to be an esteemed member of the - let say - viola group.

RADIATION RISK ASSESSMENT: CURRENT STATE AND FUTURE DIRECTIONS

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INTRODUCTION

In the context of this report, risk assessment is the process of characterizing and quantifying potential adverse health effects that may result from exposures to harmful physical or chemical agents in the environment. The concept of risk assessment is not new; it has been a factor in life faced by all living things since time immemorial. At earlier times, judgements of risk for an individual were relatively easy to make and qualitative in nature. This is because the magnitudes of most risks and their consequences were so great that they were easy to perceive and avoid. It is now becoming apparent that a totally safe or risk free society is not attainable. Everyone accepts some degree of risk as a normal part of our daily activities. Consistent with this view, legislation and executive orders have increasingly called for balancing of risks, costs and benefits rather than striving for absolute safety. This approach now requires quantitative rather than qualitative risk assessments.

The increased emphasis given to balancing risk, costs and benefits in developing regulations for factors that may influence human health has provided impetus for developing a more formalized approach to risk assessment. A U. S. National Academy of Sciences/National Research Council committee has recently reviewed risk assessment in government and offered recommendations on management of the process.[1] The approach recommended is illustrated in Figure 1. Here, risk assessment plays a central role in identifying the most important health risks that may require some degree of management and information gaps that should receive priority research consideration. The interactions between epidemiology, field measurements, assessments, risk characterization, decisions and actions are indicated.

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This report is concerned only with risk assessment and leaves risk management and the details of scientific research to other presentations at this conference. The approach taken here in reviewing risk assessment focuses mainly on radiation exposures resulting from releases of radioactive substances to the environment where they may be inhaled or ingested by people. The risk assessment process involves the series of steps illustrated in Figure 2. In the first step, the source of radioactive material must be characterized in terms of its magnitude, its potential for release, and its many physical-chemical parameters that determine its dispersion through the environment. In the second step, environmental dispersion must be projected to predict the concentrations and physical-chemical forms of radionuclides to which people may be exposed and the size of the exposed population. Thirdly, dosimetry models are used to estimate the amounts of radionuclides taken up from the environment, their accumulation in body organs, and the resulting radiation doses to tissues at risk. Lastly, dose-effect relationships are needed to estimate health risks to individuals and the total population. Each step of a risk assessment should be accomplished using the best scientific judgements. That is, calculational models should use values of parameters that can be justified on a scientific basis rather than values that result in conservative or overestimated projections of risk. Conservatism may be introduced in a later phase of risk management where the degree of conservatism can be clearly stated and recognized as a matter of judgement.

Perhaps the most desirable qualities of a risk assessment are that the methodology be credible and easily understood. To be scientifically credible, every opportunity should be taken to validate each step of the risk assessment calculations using previous measurements on the same or similar materials as they moved through the environment and were taken up by people. These data may come from studies of accidents, environmental tests, or laboratory investigations. To be credible for decision makers and the public, the risk assessment must be developed in a manner that can be communicated to scientists and nonscientists alike and be recognized as consistent with common knowledge about similar events. This is often the most difficult hurdle for scientists and the public, and one that can render scientifically valid efforts ineffective for accomplishing their goals.

NEED FOR RISK ASSESSMENTS

There are many applications for risk assessment in evaluating issues related to the development of nuclear technology today. These include (a) development of radiation exposure control guidelines, (b) development of health and environmental impact statements, (c) guidance for mitigating risk, (d) decision making in litigation, and (e) identification of research needs. The first two applications are prospective in that no injuries have yet resulted from applying the results of a risk assessment. The next two applications are generally retrospective in that risk assessment is used as a basis for decisions concerning risks that have or may have occurred in the past. It is easy to surmize that prospective risk assessments are generally less controversial and more likely to be successful in accomplishing their goals than retrospective risk assessments. The controversies often point to the need for developing acceptable criteria for performing risk assessments, especially in very controversial situations where timely decisions must be made.

The most successful uses of risk assessment have been in developing radiation exposure control guidelines for people in workplaces or the general environment. Clear examples of these are seen in the limits for exposure to ionizing radiation developed by the International Commission on Radiological Protection (ICRP).[2,3] These prospective guidelines apply to individuals and populations, and their objectives are clearly aimed at preventing injury to people who may be exposed to radiation or radioactive sources used by existing industries.

Risk assessments have also been used successfully in developing health and environmental impact statements. These may be for single facilities and events, or in a more generic sense, for newly developing technologies. Because these assessments are also prospective, there is a tendency for optimism when selecting values of risk parameters to be used for the required calculations. If the optimism is not warranted, it will soon be challenged, especially if a perception develops that there is unequal sharing of risks and benefits over the whole of the population at risk, or if an untoward accident occurs that was not properly evaluated by the original assessment.

Risk assessments have also been used for developing guidance in attempts to mitigate natural or man-made hazards. Recent examples are the evaluation of techniques for reducing exposures to radon in indoor environments and for cleaning up large areas of radioactively contaminated land. Such assessments can be used to direct the available resources toward actions to address first the most significant factors causing risk to individuals or populations.

Resolution of certain litigations may also depend upon risk assessments. This is most likely when dealing with radiation exposures that cause diseases that are not uniquely different in kind from spontaneous diseases or diseases that may be caused either by radiation or by exposures to other toxic agents in the environment. Then it becomes necessary to postulate the probable cause among several possible causes of the disease. In other cases, risk assessment has been used to determine if property was damaged in a manner that resulted in losses to current or potential future users of the property. Such debates frequently occur when radioactive substances have been released to the environment and contaminate nearby areas. Under these retrospective circumstances, every aspect of a risk assessment may be questioned and projections of risk by the contending parties are likely to differ by orders of magnitude. This situation has led many fact-finding officials to doubt seriously the scientific merit of approaches frequently used in risk assessment.

Finally, risk assessments are used to identify information gaps and determine their relative importance to societal concerns as a basis for establishing research priorities. This is an important application that links many aspects of risk management and decision making back to research as illustrated in Figure 1. Although this should not be the only basis for establishing research priorities in the health and environmental sciences, it provides one means of ranking research priorities that are aimed at very applied risk-related topics.

RADIATION EXPOSURE ASSESSMENTS

Procedures used to estimate the magnitudes of radiation exposures to people depend upon the modes of exposure and the places at which the exposures occur. As exposures become more complex and occur at more distant locations from the sources of radiation, the uncertainties in estimating exposure and dose usually increase. For exposures that occur in workplaces, external radiation monitors provide for accurate assessments of radiation dose. If the exposures include internally deposited radionuclides, then measurements of radioactivity in bioassay samples combined with mathematical models are needed to project the time pattern of dose accumulation following the exposure.[3] For exposures to the public that occur at distant locations from the sources of radiation and in uncontrolled and unmonitored areas, exposure assessments require more extensive use of environmental dispersion and dosimetry modeling. These techniques and the uncertainties involved are reviewed by other reports.[4-6]

A schematic representation of the process of estimating environmental dispersion, uptake and dose to body organs of exposed people is shown in Figure 3. This includes external irradiation and internally deposited radionuclides and has a rapidly rising acute exposure phase and a protracted exposure phase that may last for many years following the initial accidental release. The product of the exposure-dose assessment should result in estimates of the effective doses to the specific tissues and organs at risk. It is these organ doses, not the exposures, that are best related to health risk in the next step of the risk assessment. This is because dose-effect relationships are derived from previous exposures of other human populations and may involve significantly different exposure modes. However, it is critical that the doses to people for which the risk assessment is done are equivalent to the doses used in developing the dose-effect relationship.

DOSE-EFFECT RELATIONSHIPS

A central feature of risk assessments for exposures to ionizing radiation is the development of appropriate dose-effect relationships. For the purposes of this discussion, the effects of exposures to ionizing radiation can be grouped into three general categories; early effects, cancer, and genetic effects. Very high levels of acute exposure result in extensive damage to body tissues and may lead to loss of organ function and early death. These are most likely to result from external radiation exposures, but they may also involve significant internal radionuclide deposition, especially through inhalation. For example, this type of exposure occurred for people who were near the Chernobyl nuclear reactor accident and probably contributed significantly to the injuries sustained by some individuals.(7) At present, dose-effect relationships for combined exposures to external irradiation and internally deposited radionuclides must be derived from laboratory studies using animals.(8)

For radiation exposures that are not large enough to cause acute injuries and death, cancer is the most significant long-term somatic health risk. The types of cancers caused by ionizing radiaton are the same as those that occur spontaneously or as a result of exposure to chemical carcinogens. Thus, individual radiation-induced cancers cannot be identified explicitly; they can only be identified by measuring an increased incidence in a large population of irradiated individuals. This can represent a difficult task considering that total cancer mortality in most countries now exceeds 20% and some of the most common types of cancer are those considered to be sensitive to induction by ionizing radiation. Thus, for specific individuals who develop radiogenic cancers, it is only possible to estimate what portion of the individual's total calculated risk should be assigned to the radiation exposure and what portion should be assigned to other causes. This is often the subject of great controversy.

Genetic injury results from damage to reproductive cells that is not fatal to the cells or their reproductive potential. The damaged cells may produce offspring that have inherited diseases which again are similar in kind to those occurring spontaneously or as a result of exposure to chemically toxic agents. Claims of genetic injury rarely occur in radiation injury litigation even though the risk factors for predicting their occurrence are similar in magnitude to those for radiation-induced cancers.

The major sources of scientific information concerning the health effects of ionizing radiation in people are derived from studies of (a) populations exposed to nuclear weapons explosions or fallout, (b) medical patients that received radiation therapy, and (c) radiation workers.[9,10] It is important to note that these populations include many thousands of individuals, but few excess cancer deaths have actually been observed. For example, among 109,000 Japanese in the life-span study cohort of the Radiation Effects Research Foundation of Japan, only 250 excess cancer deaths were estimated to occur between 1950 and 1978 out of a total of 4756 cancer deaths.[11] Up to 1978, the total number of deaths was 23,502. Mortality from individual cancer types is shown in Table 1.

Also, in studies of over 20,000 uranium miners in the United States and Canada, only about 300 lung cancers were observed prior to 1980 while 100 spontaneous lung cancers were expected.[9] Thus, few excess cancer deaths attributable to radiation have occurred in these study populations, making it difficult or impossible to derive site specific risk factors or take into account modifying factors such as age at exposure, age at death, radiation type, genetic susceptibility, or exposures to chemical agents such as cigarette smoke.

Different types of mathematical models are applied in describing the acute effects and long-term cancer risk from exposure to ionizing radiation. Acute radiation injuries result from high doses delivered over short periods of time, and the effects are frequently seen within days or weeks. They include widespread destruction of blood-forming cells in bone marrow, cells lining the gastrointestinal tract, and cells comprising the lung, liver thyroid, and skin. The exposures causing these injuries are easy to identify because they only occur in severe accident situations. These injuries are often referred to as nonstochastic because they result from widespread cell destruction and the radiation doses must exceed a threshold level after which the severity and frequency of the effects increase with dose. The threshold type of relationship is illustrated in Figure 4.

In contrast, cancer and genetic effects of radiation are termed stochastic effects. This is because they can be initiated by injury to single cells or small groups of cells in a probabilistic sense even at very low doses. After they occur, the severity of the effect is not related to dose.

Two general mathematical models are most frequently used to represent radiation-induced cancer risk; the absolute risk model and the relative risk model (Figure 5). The absolute risk model assumes that the increased cancer risk for a given exposure (excess cases per year) begins after a latent period and continues at a constant or variable level for duration of the expression time. The latent period is the time that elapses between initiation of the cancer by the radiation dose and its clinical appearance in an individual.

The expression period is the time between the first and last appearances of cancers that are induced by any radiation dose. These concepts are also illustrated in Figure 6. The relative risk model assumes that after the latent period, excess cancer risk is a multiple of the spontaneous cancer rate over the expression time. These models are arithmetically consistent with each other when applied to a single set of epidemiologic data over the same period of observation. Both models must account for the same number of excess cancers over the study period and both will predict the same excess risk for another population if irradiated in a similar manner and studied for the same time period. However, the two model predictions may differ markedly beyond the period of time covered by their epidemiologic data base as shown by the dashed lines in Figure 5. It has not been determined which, if either, model is more appropriate. Because most cancer risks increase markedly in old age, the relative risk model often predicts 3 to 4 times more cancers will occur than the absolute risk model when lifetime projections are made.[9] This problem is especially significant in projecting risks for <u>in utero</u> irradiation which is thought to be even higher per unit dose than risks from exposures to adults. It is not known how long the higher risks of <u>in utero</u> radiation may continue during a person's lifetime.

Quantitative differences between the predictions of the absolute and relative risk models disappear when the appropriate cancer latent periods, expression times, and age sensitivity relationships become known. For leukemia and bone cancers, the latent period lasts for 2 to 5 years and the expression time lasts for about 30 years after irradiation.[9] Less is known about other cancer types which may have latent periods of 20 years or more and expression times well beyond 30 years.

One additional important difference between the relative and absolute risk models is in their predicted patterns of excess cancer risk. The relative risk model predicts that the largest portion of radiation-induced cancers will occur in people with the highest cancer risk aside from cancers caused by the irradiation. The high risk category includes people with more than average genetic susceptibility to cancer, cigarette smokers and those who are exposed to other carcinogenic agents. The absolute risk model suggests a uniform pattern of excess cancer risk in an irradiated population regardless of other risk-modifying factors. These important differences between the relative and absolute risk models can and should be investigated with further laboratory and epidemiologic studies.

Coupled with the use of an absolute or relative radiation cancer risk model is a mathematical dose-effect relationship. This relates the excess risk of developing a cancer to the amount of radiation dose received. Several examples of different mathematical relationships that have been used to relate dose to the probability of effects are also shown in Figure 4; the most common forms used are the linear and sublinear relationships. Within the dose range of any set of observations (epidemiology or laboratory animal studies), there is little or no difference between the numbers of cancers represented by each mathematical expression. However, important differences occur in the model predictions of cancer risks outside of the dose range represented by the original data. This is especially important in the low-dose region. Here, the linear function predicts a higher risk and the sublinear function predicts a lower risk. The linear function is used most frequently because it is less likely to underestimate cancer risk at low doses. It also seems to be most appropriate for high-LET radiations for which more data are available in the low dose region. Sublinear functions (i.e., linear-quadratic functions) are gaining in acceptance because they are more flexible for fitting dose-effect information over all dose ranges and the fitting process automatically adjusts for the relative importance of the linear and nonlinear components. For mixtures of low- and high-LET radiations, the linear and nonlinear functions may be combined in several ways to project the increased cancer risk.

In rare situations, the number of cancers produced per unit of dose may actually decrease with increasing dose.[12,13] This can be due to life-span shortening from diseases other than cancer, or "wasted dose" that results from cell killing, or continued accumulation of dose beyond the point at which cancer has been initiated. In any event, the dose-effect relationship may appear supralinear as shown in Figure 4.

EXTENDING THE LIMITED HUMAN DATA BASE

To this point, the discussion has focused on dose-effect relationships derived from studies of human populations. These provide the most direct information for evaluating other human exposures involving the same or very similar agents and exposure modes. Adequate human risk information is available for evaluating acute external exposures to penetrating low-LET radiation (i.e., gamma and x rays), alpha irradiation of the central respiratory airways, bone and liver, and beta-gamma irradiation of the thyroid from deposited radioiodine.

For other important types of exposures, human dose-effect information is lacking. These include inhaled alpha-emitting actinides and beta-gamma-emitting fission products, acute and protracted external exposures to neutrons and protracted external exposures to low-LET penetrating radiations. Until recently, it was thought that the exposures to the Japanese atomic bomb survivors involved a significant neutron component for which a dose-effect relationship could be developed.[9] However, it is now believed that the neutron exposure was likely to have been too small a part of the total exposure to be useful for this purpose. Hence, there are no generally accepted dose-effect relationships for neutron exposures based upon studies in humans at the present time. However, there are many laboratory studies on the effects of neutron exposures to animals, cells and subcellular systems. Thus, for neutrons and some other types and modes of radiation exposure, developing dose-effect relationships will depend upon combining results of human, animal and *in vitro* laboratory studies as suggested in Figure 7.

It is important to note that there can be significant differences between external radiation exposures and exposures that result from internally deposited radionuclides. Differences in the rates at which the dose may be delivered are illustrated in Figure 8. Here, external irradiation refers to a single acute exposure during which dose is delivered at a high rate over a very short period of time. With internally deposited radionuclides, the exposure may be brief or chronic, but the dose is almost always protracted when the biological and physical halftimes for retention are long. Because there are differences in the effectiveness of radiation for producing health effects that depend upon total dose, dose rate, age at exposure, age at death and other modifying factors, dose-effect relationships developed from information on acute radiation exposures must be modified to account for dose protraction when evaluating risk from such exposures. This can be accomplished using information derived from laboratory studies.

Whole animal studies, cell toxicity, chromosome damage and DNA transformations have also been used to determine the relative biological effectiveness (RBE) of different types of radiation. High-LET alpha radiation and neutrons were determined to be ten to twenty times more effective in producing this damage than low-LET beta and gamma radiation. In a sense, RBE is a toxicity ratio and the use of this information led to the formulation of a related system of radiation quality factors (Q) to be used for regulatory purposes when dealing with different types of radiation.[2] These are used as multipliers of the measured or calculated high-LET dose in order to obtain a new and higher effective dose that was equal in effect to the same amount of low-LET radiation. This effective dose may then be applied with low-LET dose-effect relationships to predict risk from high-LET radiations.

A second type of toxicity ratio was developed from studies in laboratory animals and people to account for differences in radiation dose distribution of radionuclides deposit in bone. In this case, toxicity ratios for bone cancer were developed by comparing the risk for radionuclides that deposited on bone surfaces with that for radionuclides that deposited throughout the bone volume. Because plutonium deposits mainly on bone surfaces near to the cells at risk, it showed a greater effectiveness in producing bone cancers than radium which deposits throughout the bone volume. This led to the use of a dose distribution factor (N) which is a multiplier for the calculated bone dose for alpha-emitting radionuclies other than radium. This effective dose is used with the dose-effect relationships developed for humans exposed to radium to predict bone cancer risk for other alpha-emitting radionuclides.

Laboratory studies using whole animals provide a means for developing dose-effect relationships for diseases such as cancer, but there are also important differences between animals and people. The most important differences are in (a) life span, (b) physiological characteristics that influence radionuclide uptake, (c) metabolic characteristics that influence clearance, and (d) sensitivity to radiation-induced cancer. Other important considerations may arise from the experimental conditions, especially if the levels of exposure used in the laboratory studies are substantially higher than those involving people.

To account for some of these differences, useful arrays of toxicity information can be constructed by combining results from laboratory and epidemiologic studies.[14,15] These may include information on the specific substance being evaluated and on related substances with similar biologic actions. The latter are referred to as surrogate substances or physical agents. Table 2 shows a sample array of data for evaluating lung cancer risk to people from inhaled particles containing alpha- or beta-gamma-emitting radionuclides. The upper row is derived from epidemiologic studies and is given in terms of the number of lung cancers expected during the lifetime of a population that received a collective radiation dose of one million rad. Radiation lung cancer risks have been measured only in people who were exposed to external x-rays, to atomic weapons explosions, and to radon and its progeny. Many studies have used laboratory animals to estimate the unknown cancer risk factors for people. The example shown in Table 2 includes studies in dogs and rats.

The lung cancer risk for people exposed to inhaled radioactive particles can be estimated from each element of the array if we know (a) the relative potency of alpha or beta radiation, compared to the other types of radiation exposures, when inhaled in particles, and (b) the relative sensitivity of people, compared to rats and dogs, to radiation-induced lung cancer. The relative potencies of the different radiation exposures can be estimated as the average ratio of elements in the first or second column of the array to those in any other column. Likewise, the relative sensitivity of people compared to laboratory animals can be estimated by the average ratio of elements in the top row of the array to the elements in any other row.

Each element of the array can then be used to provide an estimate of the human lung cancer risk from inhaled radioactive particles by multiplying it by the appropriate relative potency and relative sensitivity factors. Using this procedure, the median lifetime human lung cancer risk factor for inhaled particles containing alpha-emitting radionuclides was estimated to be 3200 lung cancers per million rad. The highest individual value is almost 7000 lung cancers per million rad, about twice the median value. Reversing the first two columns in Table 2 and repeating the calculation for beta-emitting radionuclides gives a median lifetime risk factor of 430 lung cancers per million rad, with the highest value again being about twice the median value. Distributions of these risk factors are shown in Figure 9. Further details of this use of laboratory studies are given elsewhere along with an example of its application to carcinogenic chemical exposures.[14]

APPORTIONING OF RISK FOR INDIVIDUALS

The above discussion is not intended to be a detailed treatment of dose-effect relationships used in radiation risk assessment; it is intended only to illustrate some of the complexities in evaluating risk to exposed populations. Additional difficulties arise when moving from evaluations of large populations to evaluations for individuals. Individuals have unique characteristics that may modify their risk from exposure to radiation or that may contribute to competing risks and effectively lower the probability that the radiation risk will be expressed. Perhaps the best developed example of this is the impact of cigarette smoking on the risk of radiation-induced lung cancer from inhaled radon progeny.[16]

The mathematical expression developed by Whittemore and McMillan for predicting lung cancer risk is shown in Figure 10. It employs the relative or multiplicative risk model and contains a term accounting for radiation exposure from inhaled radon progeny in units of working level months (WLM) and a term accounting for exposure to cigarette smoke in units of packs smoked over a lifetime. In the sample calculation shown in Figure 10, smoking is estimated to increase the lung cancer risk 4.7 times and exposure to radon progeny multiplies the risk 2.5 times. Even though the assumed radiation exposure would more than double the risk in a smoking miner, it only accounts for about 20% of the total risk. The cigarette smoke exposure alone accounts for about 40% of the total risk, so that the remaining 40% of the risk is related to interactive factors. It is difficult to recommend how this type of information should be used to apportion risk among smoking and radiation; however, this is a question of policy rather than of science.

In the Orphan Drug Act of 1983, the U. S. Congress requested the Secretary of Health and Human Services to publish tables of information to estimate the likelihood that cancers developed by people were caused by a previous radiation exposure.[17] The original concern focused on thyroid cancers developed by people exposed to radioiodine in nuclear weapons fallout, but the mandate extends beyond this to other radiation-induced cancers. Similar calculations have been used in radiation-injury litigations where it was necessary to present testimony on whether a cancer developed by an individual was, more likely than not, caused by a previous exposure to radiation. This application is now commonly referred to as a calculation of the assigned share of causation attributable to radiation.

The assigned share attributable to radiation has been estimated by using either the absolute or relative risk model calculations. Using the absolute model, the assigned share attributable to the radiation exposure is expressed as;

$$\text{Radiation Assigned Share} = \frac{\begin{array}{c}\text{Excess Cancers per}\\ \text{Unit Population per}\\ \text{Unit Dose}\end{array} \times \text{Dose}}{\begin{array}{c}\text{Expected Cancers}\\ \text{per Unit Population}\end{array} + \begin{array}{c}\text{Excess Cancers per}\\ \text{Unit Population per}\\ \text{Unit Dose}\end{array} \times \text{Dose}}$$

The expected cancer risk should include all lifestyle factors that contribute to the development of cancer in an individual, although many such factors are already accounted for in an average sense by the epidemiological data bases. Special consideration should be given to hereditary factors, exposures to medical radiation and other carcinogenic agents, age, location of residence, smoking and dietary factors. Using the relative cancer risk model, fractional causation is expressed as;

$$\text{Radiation Assigned Share} = \frac{\begin{array}{c}\text{Fractional Increase}\\ \text{in Risk per Unit Dose}\end{array} \times \text{Dose}}{1 + \begin{array}{c}\text{Fractional Increase}\\ \text{in Risk per Unit Dose}\end{array} \times \text{Dose}}$$

This model assumes that radiation simply multiplies an individual's risk of developing a cancer due to all other causes whatever they may be. When the assigned share from radiation exceeds 0.5, using either model calculation, it may be argued that the cancer was, more likely than not, attributable to the radiation exposure.

The radiation cancer risk tables recently provided to Congress by the Secretary of Health and Human Services recommend use of the relative (multiplicative) risk model calculation in evaluating all situations except for lung cancer developed by smokers.[18] However, the absolute risk model calculation is difficult to use for this purpose because its use requires specific lifestyle information for an exposed individual which does not lend itself to presentation in simple tables. The relative risk model calculation is simpler to use but it has not been demonstrated that it can be applied to radiation risk calculations regardless of an individual's previous exposure to other carcinogenic agents. It is also unclear as to how cancer risks from multiple agents might be incorporated into the relationships for calculating assigned share. Even if the assumptions of the relative risk model can be demonstrated, people who have lifestyle factors that increase their spontaneous risk of developing cancer are contributing to the increased probability that a cancer will actually occur. As a consequence of the assigned share approach, employers might well avoid placing smokers, women and individuals over 40 years of age in jobs that involve exposures to radiation.

Such a policy might have a major impact on employment practices with little or no scientific justification. It might also constitute a new way of establishing <u>de facto</u> occupational radiation exposure controls, because employers could avoid litigations over radiation-induced cancers by not allowing their employees to accumulate more dose than could be conceived of as doubling the spontaneous risk of developing the most radiation sensitive types of cancer. Unfortunately, the basic uncertainty in estimating radiation cancer risk is considerable, and the whole process of calculating the assigned share of the cancer risk attributable to radiation may be too imprecise to be used in resolving most litigations.

The approach using assigned share was subsequently reviewed by a committee of the U. S. National Research Council.[19] They expressed concern that the calculation of assigned shares for radiation sensitive cancers was complex and not clear for all situations. They also pointed to the need for developing estimates of uncertainty related to the assigned shares because this may place clear limitations on their use in resolving radiation injury litigations.

FUTURE DIRECTIONS IN RISK ASSESSMENT

Several important information gaps must be filled in the near future if risk assessment is to play a more significant role in resolving public concerns and debates over issues of radiation risk. These include the development of new information in the areas of:

1. Projecting human health risks from exposures to neutrons and a wider range of internally deposited radionuclides,
2. Projecting interactions between different types of exposures to ionizing radiations and combinations of ionizing radiations and chemically toxic substances.
3. Validating appropriate mathematical models for projecting health risks from acute and chronic exposures to radiation, and

4. Developing generally acceptable guidelines for performing risk assessments and estimating assigned shares for radiation as a cause of cancer.

Although this report focused mainly on problems associated with performing risk assessments for accidental or potential exposures to radiation, and it might appear to be a very difficult or impossible task to perform in a scientifically defensible manner, the alternatives to risk assessment are unappealing. Scientists and politicians should generally recognize the strengths and weaknesses of such approaches, and then agree on a reasonable set of criteria that would be acceptable for applying risk assessment in each of the areas described at the beginning of this report.

REFERENCES

1. National Research Council Committee on the Institutional Means for Assessment of Risks to Public Health. Risk Assessment in the Federal Government; Managing the Process, National Academy Press, Washington, DC, 1983.

2. International Commission on Radiological Protection. Recommendations of the International Commission on Radiological Protection, ICRP Publication 26, Pergamon Press, Oxford, 1977.

3. International Commission on Radiological Protection. Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, Pergamon Press, Oxford, 1979.

4. U. S. Department of Energy Office of Nuclear Energy. Models and Parameters for Environmental Radiological Assessments (C. W. Miller, ed.), Technical Information Center, U. S. Department of Energy, Washington, DC, 1984.

5. Little, C. A. and C. W. Miller. The Uncertainty Associated with Selected Environmental Transport Models, Oak Ridge National Laboratory Report, ORNL 5528, 1979.

6. Cuddihy, R. G., R. O. McClellan and W. C. Griffith. Variability in Target Organ Deposition Among Individuals Exposed to Toxic Substances, Toxicol. Appl. Pharmacol. 49: 179-187, 1979.

7. USSR State Committee on the Utilization of Atomic Energy. The Accident at the Chernobyl Nuclear Power Plant and its Consequences, Information Compiled for the IAEA Experts Meeting, Vienna, August, 1986.

8. Scott, B. R. and F. F. Hahn. Early Occurring and Continuing Effects. In Health Effects Model for Nuclear Power Plant Accident Consequence Analysis (compiled and edited by J. S. Evans, D. W. Moeller and D. W. Cooper), NUREG/CR-4214, 1985.

9. Committee on the Biological Effects of Ionizing Radiations, National Research Council. The Effects of Populations of Exposure to Low Levels of Ionizing Radiation: 1980, National Academy Press, Washington, DC, 1980.

10. United National Scientific Committee on the Effects of Atomic Radiation (UNSCEAR). Sources and Effects of Ionizing Radiation, United Nations, New York, 1977.

11. Kato, H. and W. J. Schull. Studies of the Mortality of A-Bomb Survivors; 7. Mortality, 1950-1978: Part 1 Cancer Mortality, Radiat. Res. 90: 395-432, 1982.

12. Cuddihy, R. G. Risks of Radiation Induced Lung Cancer. In Critical Issues in Setting Radiation Dose Limits, NCRP Proceedings No. 3, pp. 133-152, 1982.

13. National Council on Radiation Protection and Measurements. Evaluation of Occupational and Environmental Exposures to Radon and Radon Daughters in the United States, NCRP Report No. 78, 1984.

14. Cuddihy, R. G., B. B. Boecker, F. F. Hahn and R. O. McClellan. Human Risk Relationships Derived from Epidemiology and Laboratory Studies, Inhalation Toxicology Research Institute Annual Report, LMF 107, pp. 363-371, 1983.

15. DuMouchel, W. H. and J. E. Harris. Bayes and Empirical Bayes Methods for Combining Cancer Experiments in Man and Other Species, Department of Economics, Massachusetts Institute of Technology, Technical Report No. 24, 1981.

16. Whittemore, A. S. and A. McMillan. Lung Cancer Mortality Among U. S. Uranium Miners; A Reapprisal, J. Nat. Can. Inst. 71: 489, 1983.

17. Orphan Drug Act, Public Law No. 97-414, Sec. 7, 96 Stat 2059, 1983.

18. U. S. Department of Health and Human Services. Report of the National Institutes of Health Ad Hoc Working Group to Develop Radioepidemiological Tables, NIH Publication No. 85-2748, 1985.

19. U. S. National Research Council, Oversight Committee on Radio-epidemiologic Tables. Assigned Share for Radiation as a Cause of Cancer, National Academy Press, Washington, DC, 1984.

20. McClellan, R. O. Health Effects from Internally Deposited Radionuclides Released in Nuclear Disasters, in the Control of Exposure of the Public to Ionizing Radiation in the Event of Accident or Attack, Proceedings of a Symposium Sponsored by the National Council on Radiation Protection and Measurements held in Reston, Virginia, April 27-29, 1981.

21. McClellan, R. O. Health Effects of Diesel Exhaust: A Case Study in Risk Assessment, Am. Ind. Hyg. Assoc. J. 47: 1-13, 1986.

Table 1

Summary of Cancer Deaths by Site Between 1950 and 1978 in Japanese Extended Life-Span Study Sample Conducted by the Radiation Effects Research Foundation[11]

Cancer Type	Total Deaths	Excess Deaths
Leukemia	180	91 (50)[a]
All cancers except leukemia	4576	160 (3.5)
Esophagus	156	8 (5.1)
Stomach	1754	42 (2.4)
Colon	157	16 (10)
Other digestive	595	24 (4.0)
Lung	459	32 (7.0)
Breast (female)	128	15 (12)
Urinary tract	104	8 (7.7)
Mutliple myeloma	20	6 (30)
All Causes	23502	

[a]Figures in parentheses are the excess cancer deaths expressed as percentages of the total cancer deaths by site.

Table 2

Summary of Lifetime Lung Cancer Risks in People and Laboratory Animals Exposed to External Radiation and Inhaled Radioactivity

	Lung Cancers per Million Rad			
	Inhaled Particles		External	Radon and
	Alpha	Beta	X-rays	Daughters
People	?[a]	?	140 (100)[b]	1000
Rats	1750	220	130	1500
Dogs	631	80	60	60

[a]Unknown value to be estimated from other elements in the Table, as described in text.

[b]Number in parentheses refers to Japanese atomic bomb survivors exposed mainly to gamma radiation.

FIGURE TITLES

Figure 1. Schematic representation of the relationships between scientific research, risk assessment and risk management. Adapted from a report of the U. S. National Academy of Sciences/National Research Council.[1]

Figure 2. Outline of steps used in assessing human exposures, radiation doses and health risks resulting from releases of radioactive materials to the environment.

Figure 3. Time relationships relating levels of radioactivity in the environment, uptake or external exposures to people, dose to tissues at risk, and potential health effects. Adapted from McClellan _et_ _al_.[20]

Figure 4. Comparison of different mathematical forms of dose-effect relationships emphasizing extrapolations from the high dose region where effects information is available to the low dose region where no information may be available.

Figure 5. Patterns of excess cancer risk predicted for exposures to ionizing radiation using the constant risk, absolute and relative risk models.

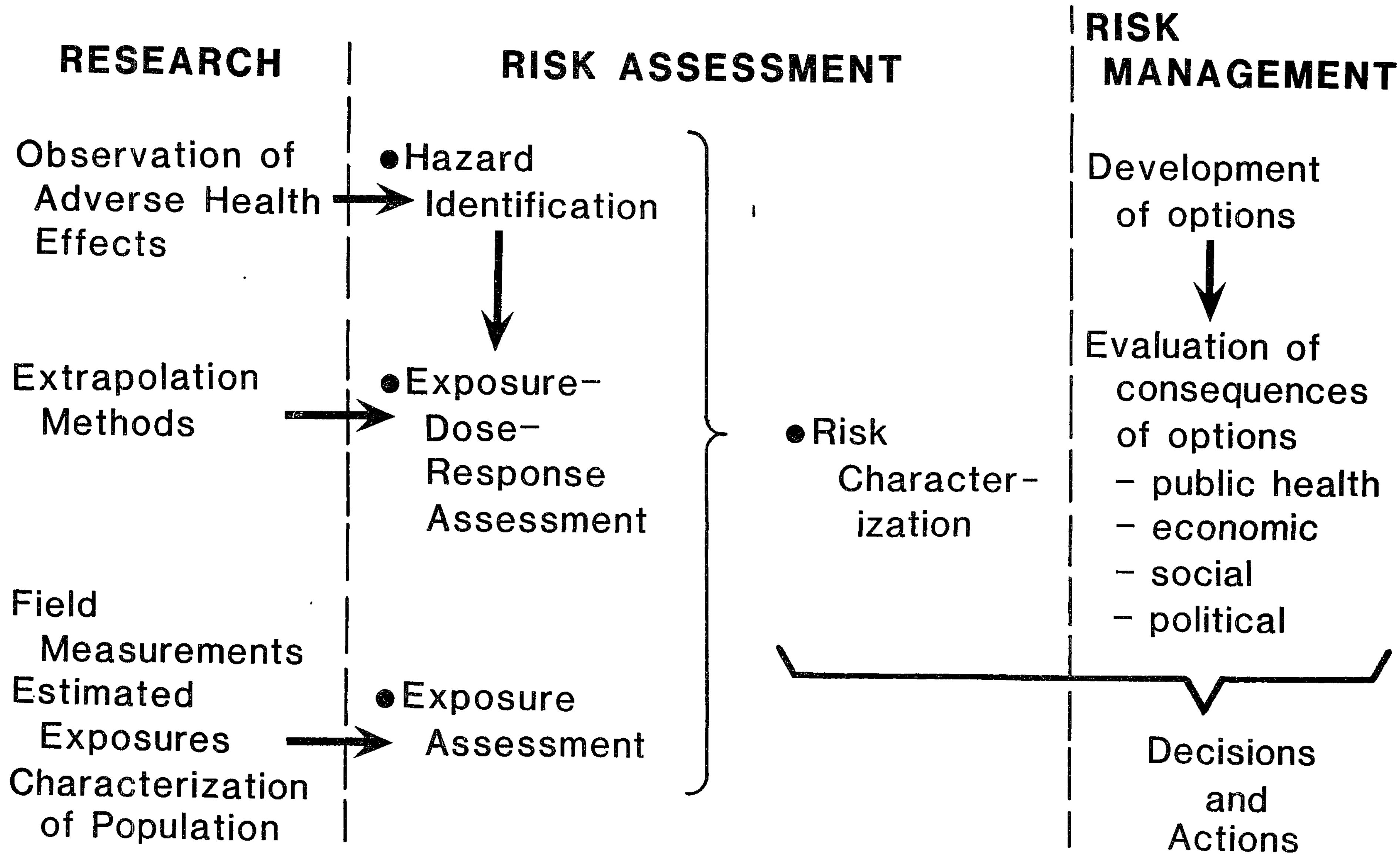
Figure 6. Illustration of the time relationship between cancer induction, latent period, and cancer expression. Also shown is the level of biological change as the health effect progresses.

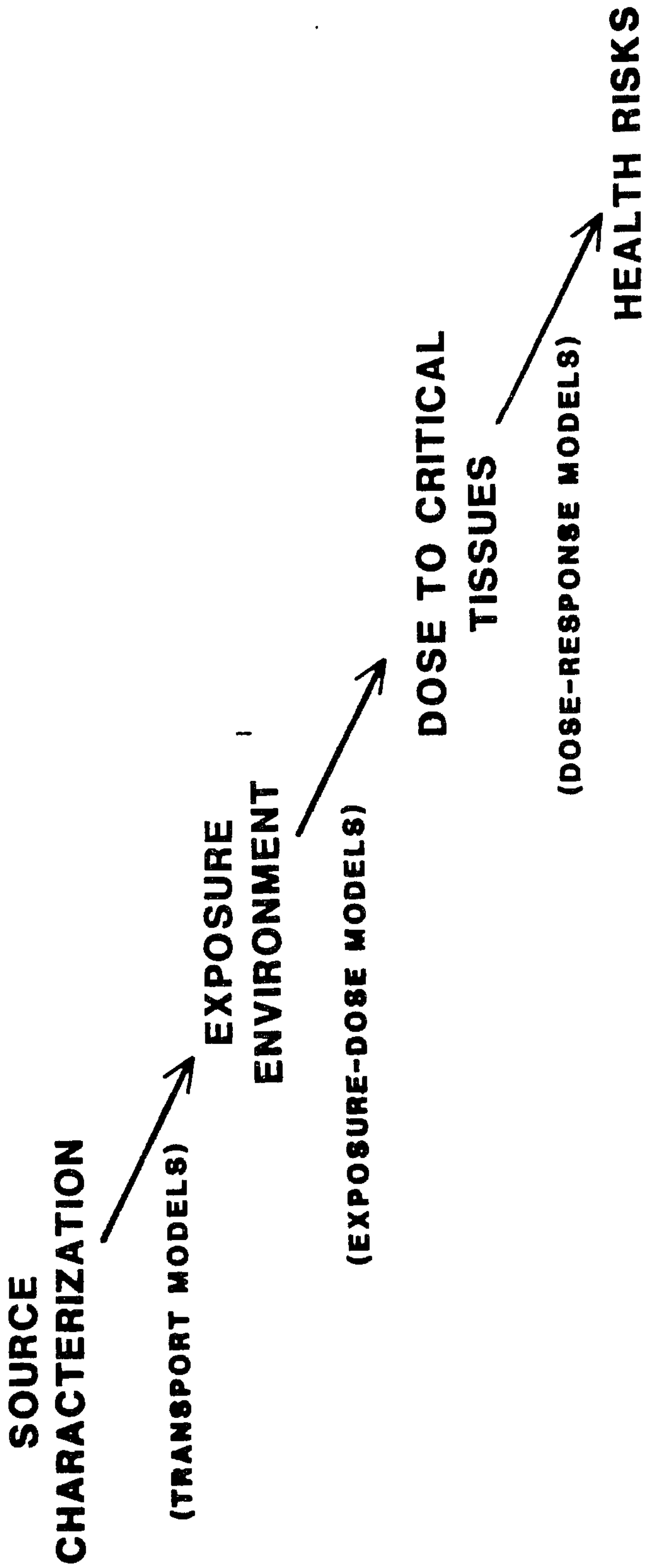
Figure 7. Multiple sources of information are often required in performing risk assessments. Illustration reproduced from McClellan.[21]

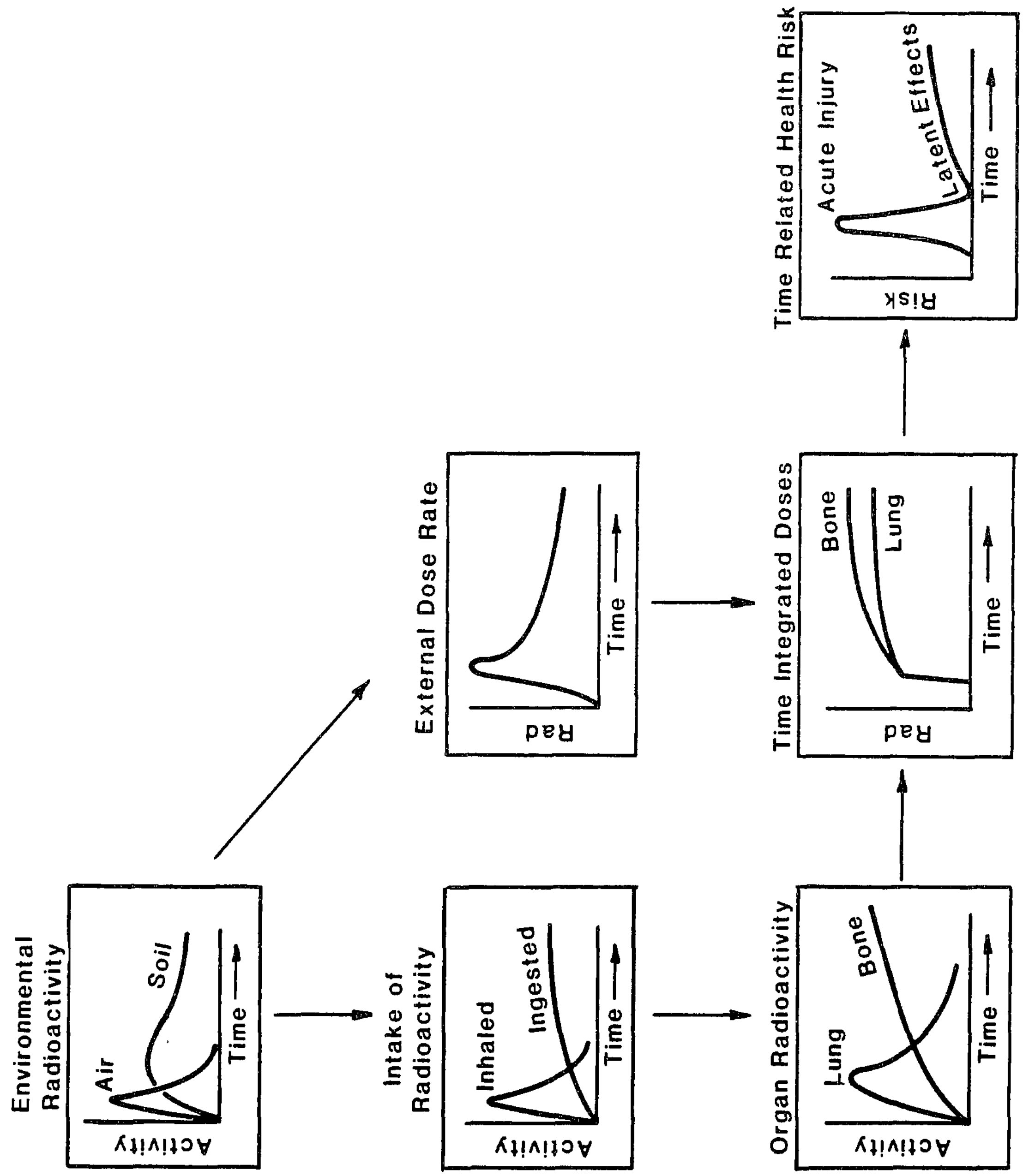
Figure 8. Summary of time relationships between exposure, dose and age for acute external irradiation and internally deposited radionuclides. The majority of data on human health effects are derived from studies involving external irradiation, and very little data is available for internal radionuclides.

Figure 9. Distribution of calculated lung cancer risk factors for people who inhale particles containing alpha- and beta-gamma-emitting radionuclides. Calculation involves data shown in Table 2.

Figure 10. Sample lung cancer risk calculation for a cigarette smoker who is also exposed to inhaled radon progeny. Calculation uses the equation of Whittemore and McMillan.[16]


RESEARCH
RISK ASSESSMENT
RISK MANAGEMENT
Observation of Adverse Health Effects
Extrapolation Methods
Field Measurements
Estimated Exposures
Characterization of Population
Hazard Identification
Exposure–Dose–Response Assessment
Exposure Assessment
Risk Character-ization
Development of options
Evaluation of consequences of options
– public health
– economic
– social
– political
Decisions and Actions


SOURCE CHARACTERIZATION
(TRANSPORT MODELS)
EXPOSURE ENVIRONMENT
(EXPOSURE-DOSE MODELS)
DOSE TO CRITICAL TISSUES
(DOSE-RESPONSE MODELS)
HEALTH RISKS


Environmental Radioactivity
Air
Soil
Activity
Time
Intake of Radioactivity
Inhaled
Ingested
Activity
Time
Organ Radioactivity
Lung
Bone
Activity
Time
External Dose Rate
Rad
Time
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Lung
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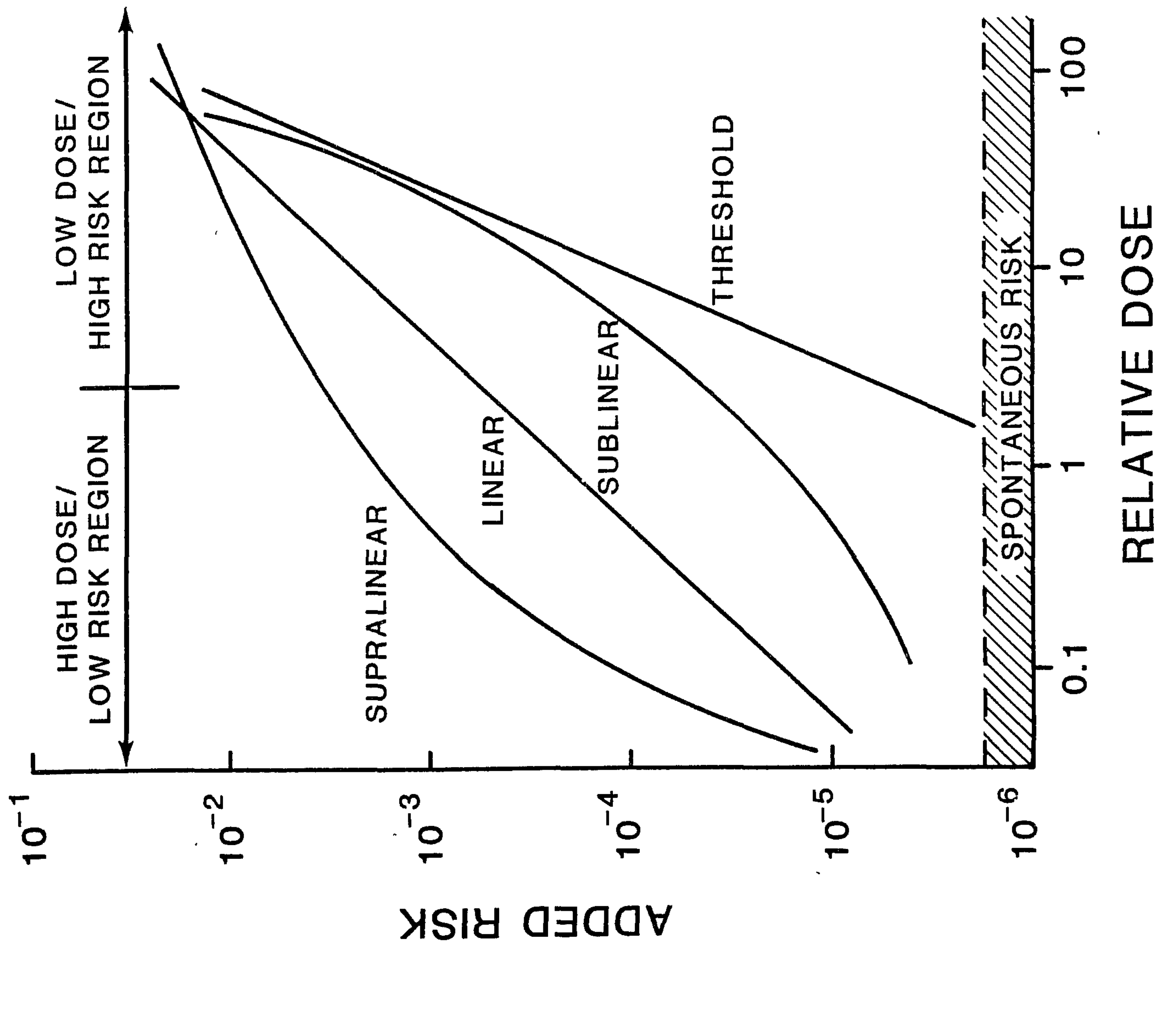
Time
Time Related Health Risk
Acute Injury
Latent Effects
Risk
Time


HIGH DOSE/
LOW RISK REGION
LOW DOSE/
HIGH RISK REGION
SUPRALINEAR
LINEAR
SUBLINEAR
THRESHOLD
SPONTANEOUS RISK
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
10^{-6}
ADDED RISK
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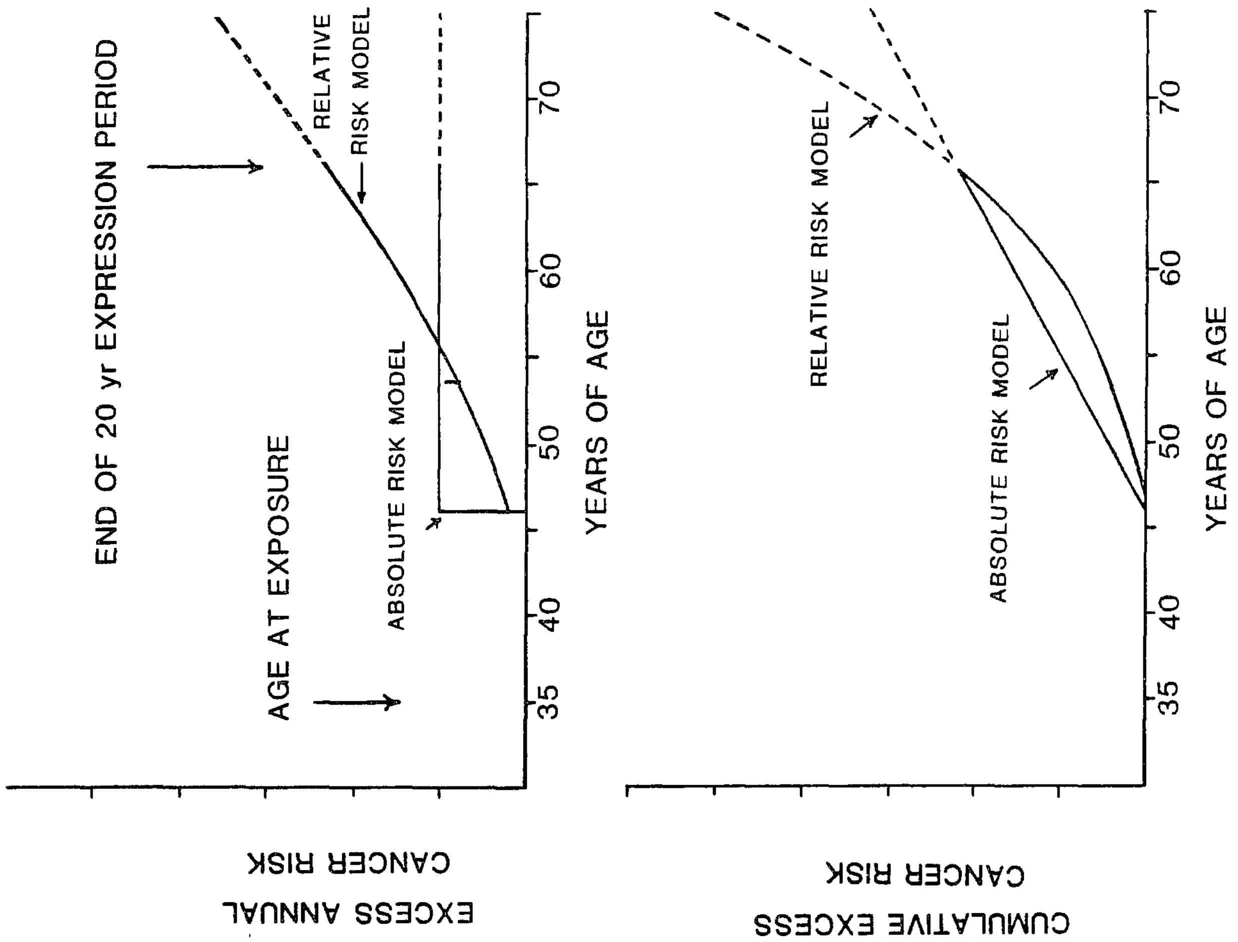
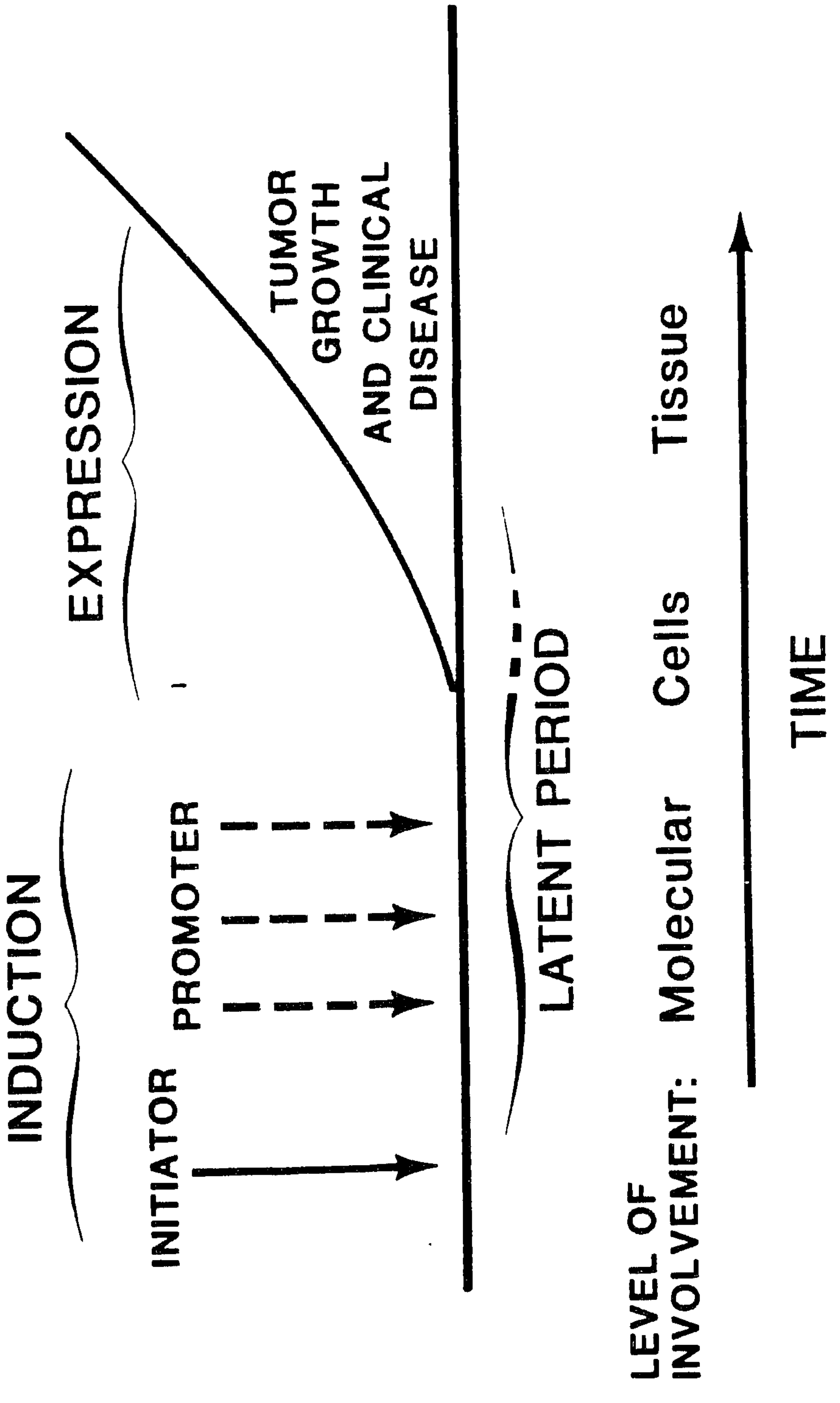
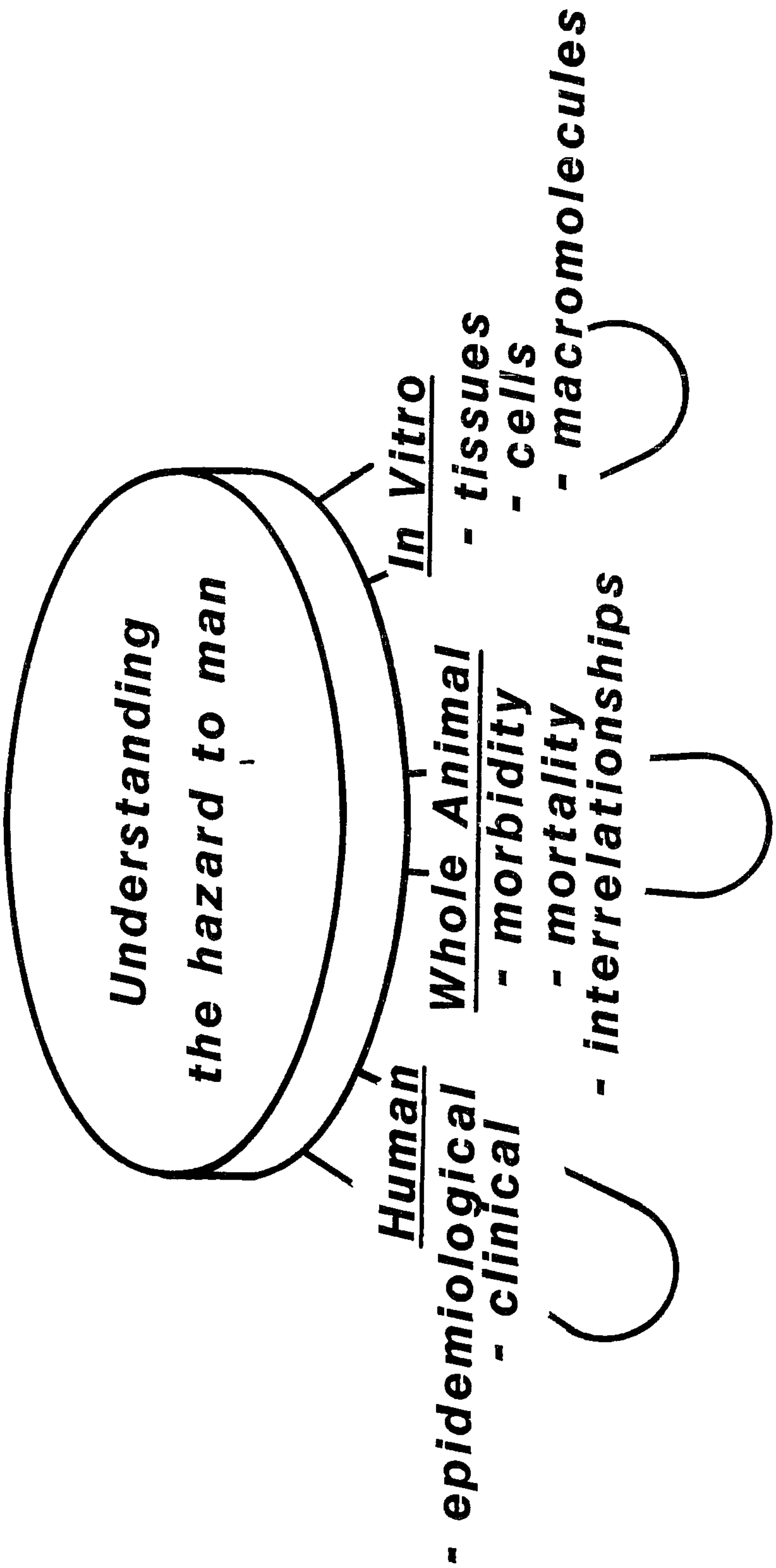
1
10
100
RELATIVE DOSE


END OF 20 yr EXPRESSION PERIOD
AGE AT EXPOSURE
RELATIVE RISK MODEL
ABSOLUTE RISK MODEL
35
40
50
60
70
YEARS OF AGE
EXCESS ANNUAL CANCER RISK
RELATIVE RISK MODEL
ABSOLUTE RISK MODEL
35
40
50
60
70
YEARS OF AGE
CUMULATIVE EXCESS CANCER RISK


INDUCTION
EXPRESSION
INITIATOR
PROMOTER
TUMOR GROWTH AND CLINICAL DISEASE
LATENT PERIOD
LEVEL OF INVOLVEMENT:
Molecular
Cells
Tissue
TIME

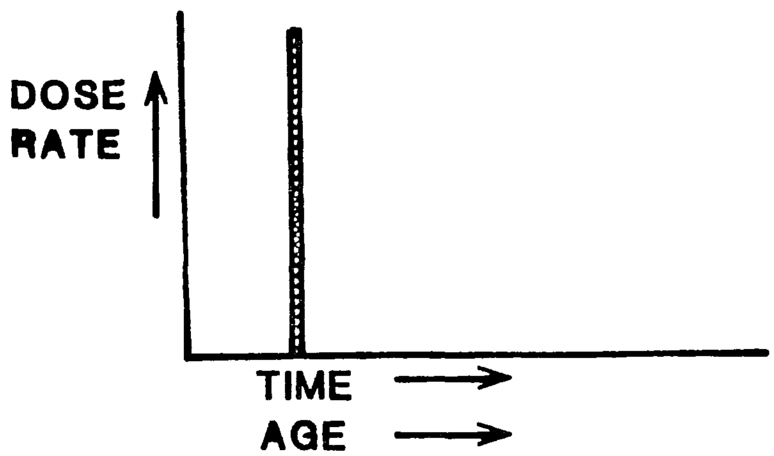

Understanding
the hazard to man
In Vitro
- tissues
- cells
- macromolecules
Whole Animal
- morbidity
- mortality
- interrelationships
Human
- epidemiological
- clinical

EXTERNAL IRRADIATION

EXPOSURE AND DOSE OCCUR AT SAME TIME

DOSE - RELATIVELY UNIFORM

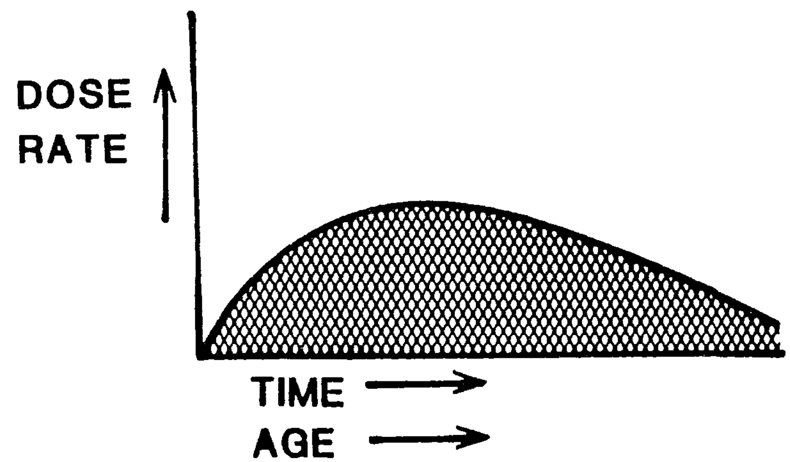
MAJORITY OF DATA ON EXPOSED PEOPLE

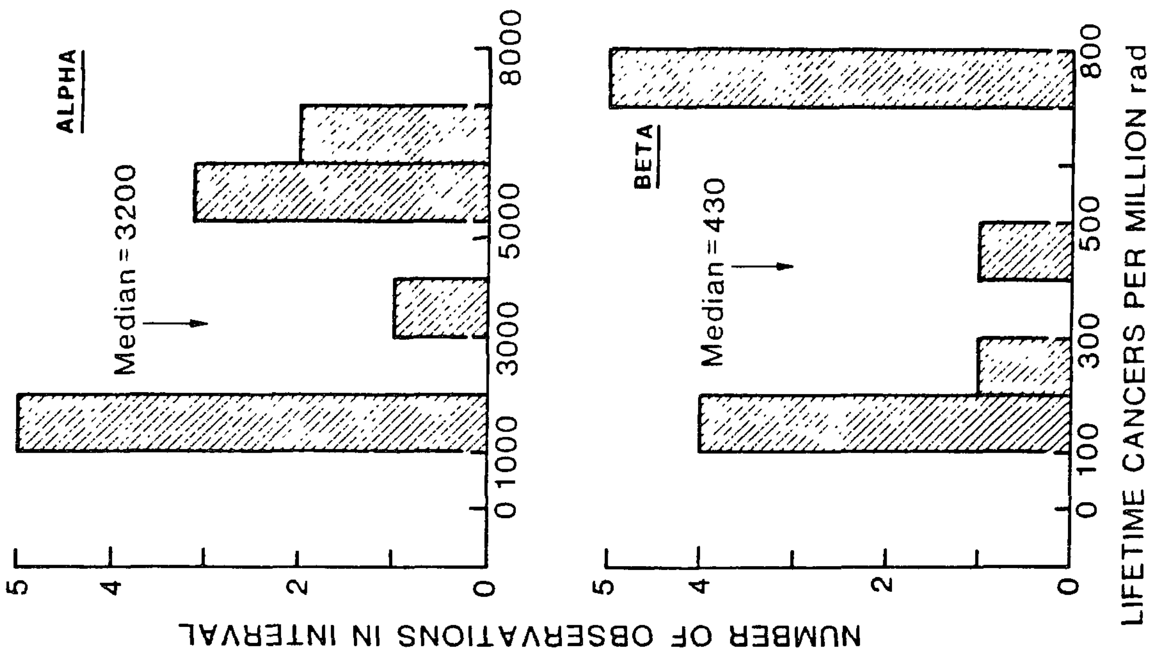


RADIONUCLIDES

EXPOSURE - MAY BE CHRONIC

DOSE - VERY LIKELY WILL BE CHRONIC AND NONUNIFORM




ALPHA
Median = 3200
BETA
Median = 430
NUMBER OF OBSERVATIONS IN INTERVAL
LIFETIME CANCERS PER MILLION rad
0
2
4
5
0 1000
3000
5000
8000
0 100
300
500
800

LUNG CANCER = (1 + 0.0031 X WLM)(1 + 0.00051 X PACKS)

ASSUME: 500 WLM EXPOSURE TO RADON DAUGHTERS
7300 PACKS EXPOSURE TO CIGARETTES

RELATIVE RISK RELATIONSHIP

		RADON RISK	
		NO	YES
SMOKING RISK	NO	1 →	2.5 (20%)
	YES	4.7 (40%) →	12 (100%)

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RESEARCH AND DEVELOPMENT AT THE INSTITUTE OF RADIOLOGICAL PROTECTION AND THE ENVIRONMENT (PRYMA INSTITUTE) OF THE CENTRE FOR ENVIRONMENTAL AND TECHNOLOGICAL ENERGY RESEARCH (CIEMAT)

(Spain)

Introduction

The PRYMA Institute is responsible within CIEMAT for a programme in the area of energy-environment interaction.

The basic objective of this programme is to optimize the compatibility space between industrial activity and environmental quality or, in simpler terms, between productivity and regulation.

The emphasis of the programme is on facilitating better compliance by nuclear installations with the environmental regulations affecting them. This should, as a matter of course, lead to greater protection of the public and its environment.

Regulatory compliance is supported through activities aimed at optimizing standards and operations, promoting a better understanding of the environment and the way it responds to disturbances, making available effective measurement methods, preparing for accident situations, and contributing to personnel training.

Lines of action

The work programme of the PRYMA Institute is organized around five lines of action:

- Environmental behaviour of long-lived radionuclides;

- Environmental impact of nuclear energy;

- Internal radiological protection;

- Atmospheric physico-chemistry and the effects of contaminants;

- Biological effects of environmental aggressors.

The first three lines together represent a radiological protection subprogramme, while the last two are concerned with subjects relating to conventional contaminants, without excluding those of a radioactive nature.

The various lines are discussed below.

Environmental behaviour of long-lived radionuclides

The activities of this project pursue two objectives: on the one hand, to gain a better understanding of the environmental behaviour of radionuclides that, because of their long life and toxicity, are of importance with regard to the management of radioactive waste, and on the other, to meet the national social requirements for radioactive monitoring in zones in which certain of these radionuclides are located. Both objectives are complementary, the effect of the studies being to produce better monitoring. In addition, this line includes the question of the strategies to be followed for the reclamation of agricultural areas following radionuclide contamination.

This line of work is structured around three subprojects:

- Radiological follow-up and monitoring of the transuranic contamination at Palomares (Almería), which involves the study of the behaviour of plutonium and americium in a land environment, with the co-operation of the United States Department of Energy (DOE) and a number of American laboratories and in conjunction with the CSN in Spain;

- Behaviour of plutonium and americium in a marine environment within the framework of the European Economic Community's radiological protection R and D programme;

- Strategies for the recovery of agricultural soil after radioactive contamination, in the wake of the Chernobil accident and in co-operation with various European laboratories (RISØ of Denmark, RIVM of the Netherlands, CEA of France) under the auspices of the EEC.

Environmental impact of nuclear energy

This line covers all the environmental subjects in the area of radiological protection having to do with better mission compliance on the part of the nuclear fuel cycle installations of the related organizations - ENUSA, CC NN, ENRESA and CSN. It also involves collaboration with regional and local authorities and in monitoring and specific studies.

This line of work is structured around three subprojects:

- Radiological impact in the final disposal of high-activity radioactive waste; modelling problems, determination of parameters and scenarios, criteria analysis and international follow-up of the subject; all of this as support for the management of ENRESA;

- Radiological monitoring plans and studies, designs and data evaluation in support of operations for all fuel cycle installations and in support of the authorities;

- Preparation for emergencies through the development and adaptation of ad hoc procedures, with specific reference to the operation of mobile facilities.

Internal radiological protection

This line includes the subjects of operational radiological protection and protection of exposed personnel, according to the following main subprojects:

- Radiological protection methodology in the operation, dismantling and shutting down of installations, where in addition to the support provided for on-stream installations particular attention is directed to the acquisition of experience in the work of dismantling, taking advantage for this purpose of the experience gained in the reconversion and closure of a number of CIEMAT installations;

- Studies on indoor radiological protection and on external and internal personal dosimetry.

Atmospheric physico-chemistry and contamination effects

Although this project is generally concerned with energy generation and consumption, it also covers atmospheric studies at radioactive nuclear installations.

This line is philosophically based on the importance of mesometeorological phenomena decisively influenced by local conditions. This concept is essential for the modernization of the atmospheric dispersion approximations that have thus far been used in the nuclear field and whose validity is more than doubtful.

Biological effects of environmental aggressors

This line also is generally concerned with all the aggressors, but includes radiobiology. It specifically includes the effects on the haematopoietic system with particular attention to two phenomena:

- The existence of radio-induced humoral factors that modify the dynamics of the system;

- The existence of glucopeptides that modify the response of the system to irradiation, with the hope that they will make possible preventive or corrective interventions.

In addition, the subject is considered at the molecular level through a study of mutagenesis and genetic expression.

RADIATION PROTECTION IN BRAZIL

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(Brazil)

INTRODUCTION

As presented in the first session of Committee II, Brazil has chosen to prepare for nuclear power generation because of its future electrical needs and availability of large uranium reserves. The success of a nuclear programme depends not only on the mastering of technology but mainly on safety assurance. Brazil also makes extensive use of nuclear energy in medicine and industry, as described yesterday in the third session of Committee II. Therefore, a Radiation Protection Programme has been created within the field of nuclear safety.

Such programme branches out into two, as follows:

- a regulatory system, coherent with international recommendations and regulations such as those from ICRP, IAEA, ISO, etc, and
- an adequate technical and scientific structure to support it.

The first system has, at its top, the country's regulatory body for nuclear energy, the Comissão Nacional de Energia Nuclear (CNEN). Its workings on this field will be presented on Tuesday, during the sixth session of Committee II. The second branch is mainly represented by the activities of the Instituto de Radioproteção e Dosimetria (IRD) of CNEN. IRD is responsible for radiation protection of man and the environment with strong support from ionising radiation metrology. The latter assures that radiation measurements are reliable and uniform throughout the country and that they are traceable to the international metrological system.

IRD belongs to the Secondary Standard Dosimetry Laboratory (SSDL) network of IAEA-WHO and its metrology department is in the last stages towards being officially recognised as the National Laboratory for Ionising Radiation Metrology. The SSDL/ Rio de Janeiro supervises two other regional laboratories with whose help it meets the country's demand for calibration. An extensive quality control programme, involving internal, national and international intercomparisons, has been implemented to assure the accuracy of calibration factors, radiation exposures and the activity of reference radionuclide sources furnished to users.

Since a complete report on radiation protection activities would be far too long, only a few examples will be discussed.

ENVIRONMENTAL PROTECTION

Within the licensing process of nuclear power plants and fuel cycle industrial installations, the applicant must present an environmental programme which has to be carried out in the pre-operational phase and then, routinely, during the operational phase. Simultaneously, CNEN performs an independent check programme through IRD.

The IRD/CNEN pre-operational environmental monitoring programme for the Angra NPP site, started in September 1979, will be presented as an example.

Figure 1 shows the location of the NPP site and the distribution of the thermoluminescent dosimeter and ionisation chamber measuring stations, which were used to measure radiation directly. Results are presented in figures 2 to 4.

The concentration levels of natural and artificial radionuclides were determined in primary media like air, surface and sea water, in accumulators such as soil, sand and sediments, in indicators such as grass and seaweed, and in the main local produce which includes marine produce (fish, shellfish and shrimps), milk, bananas, manioc and oranges. Sampling points are shown in figure 5. The measurements included gross alpha and beta activity measurements and gamma spectrometry. Specific techniques were used for the measurement of the activity concentration of tritium

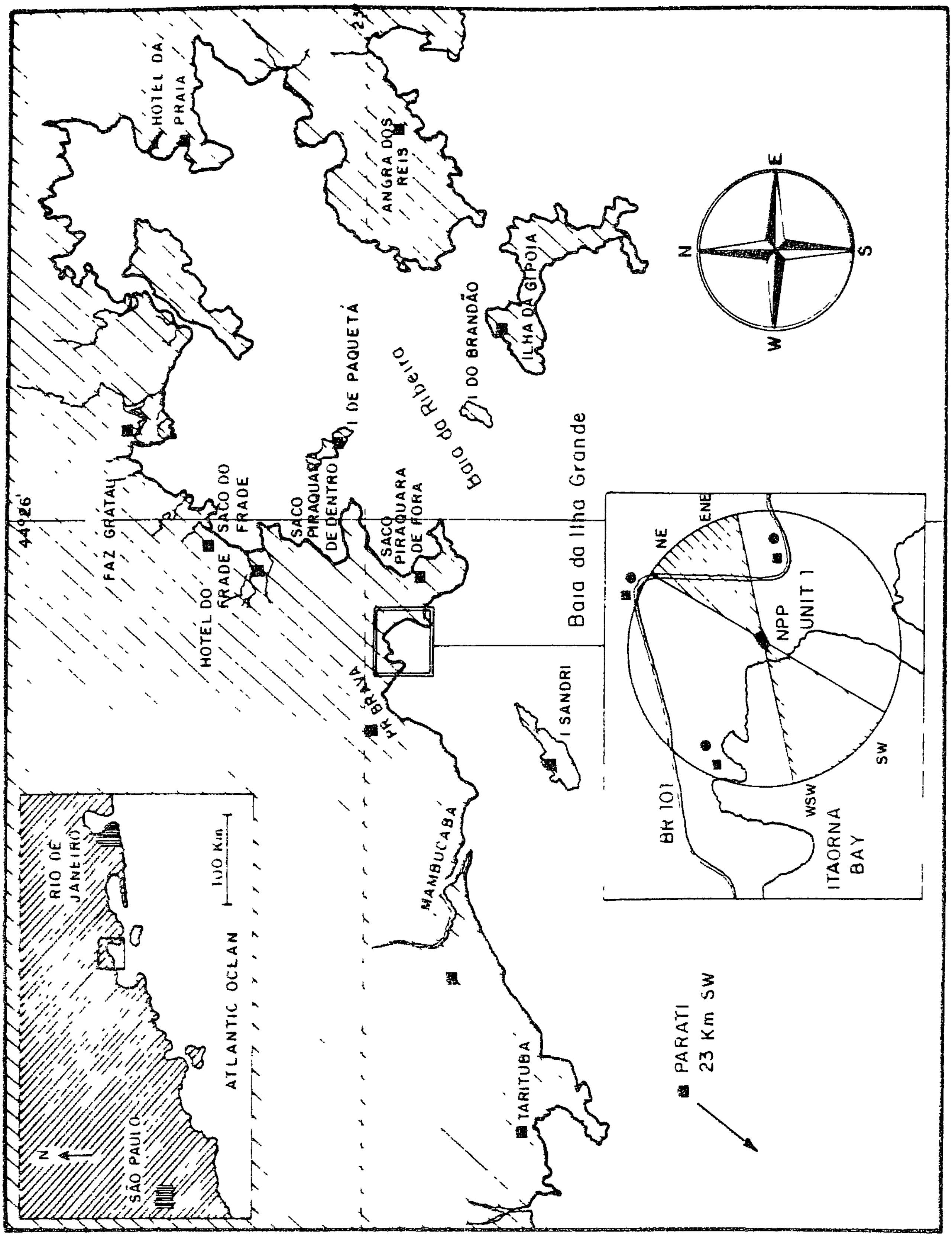


Fig. 1: Geographical location of the Angra Nuclear Power Plant Site and distribution of TLD (■) and ionization chamber (●) measurement stations.

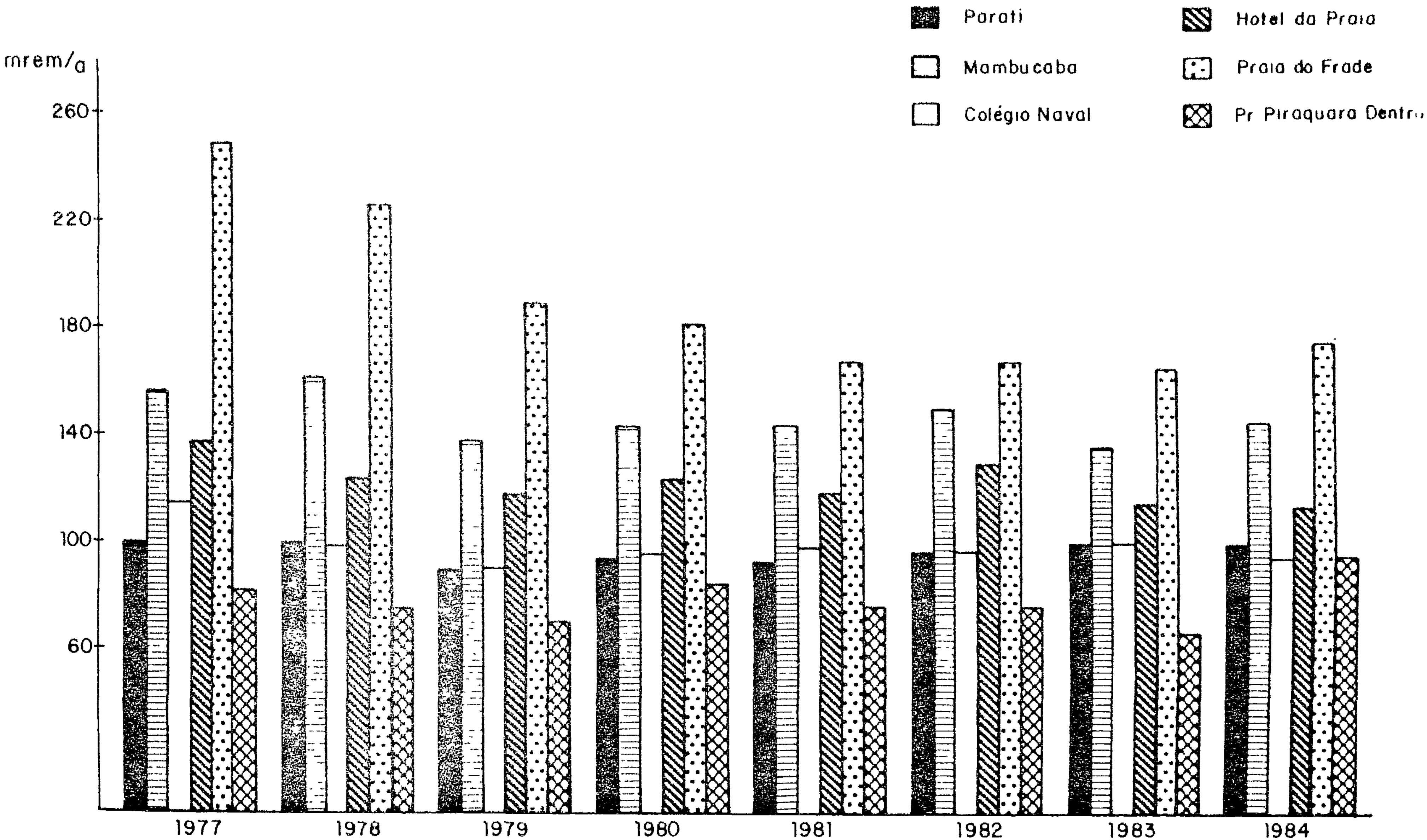


Fig.2 : Yearly $CaSO_4$-TLD results for six measurement stations from 1977 to 1984.

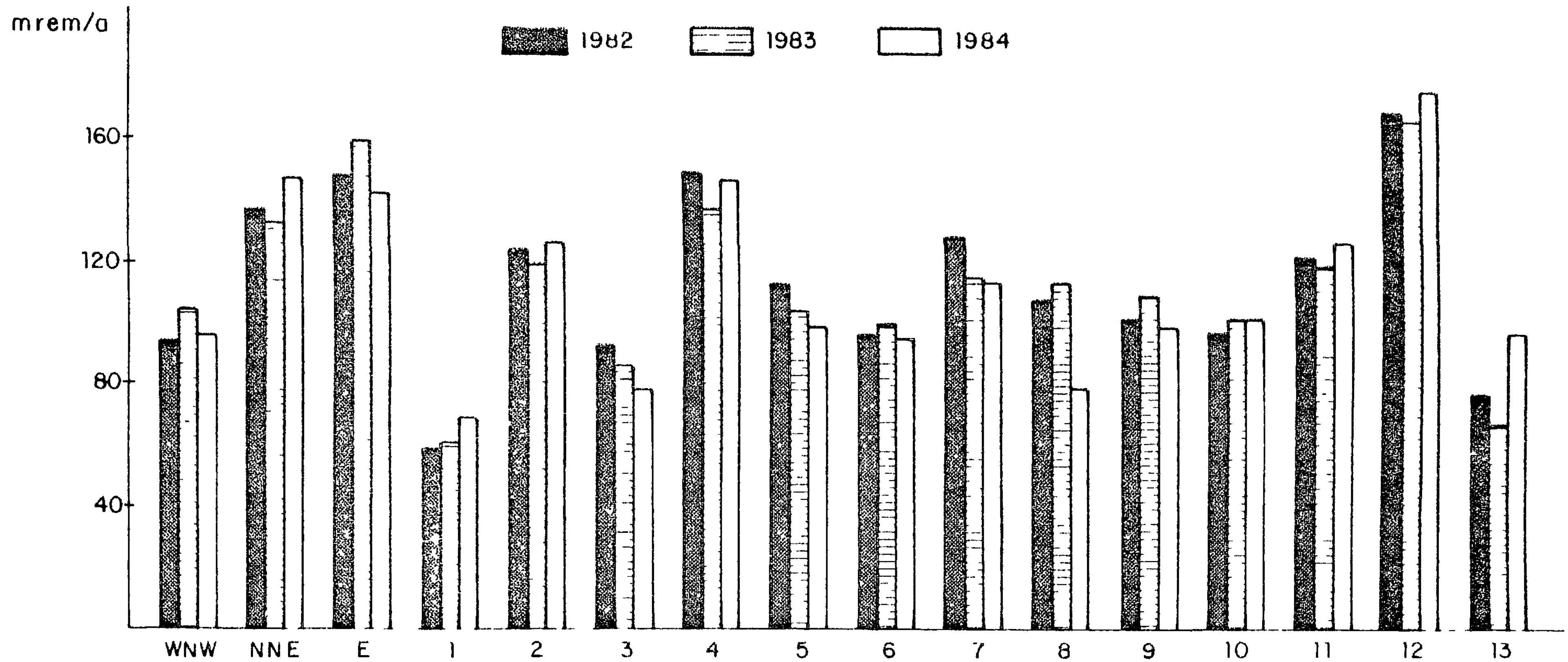


Fig 3 Yearly $CaSO_4$-TLD results for all measurement stations from 1982 to 1984 The stations are numbered according to the program description

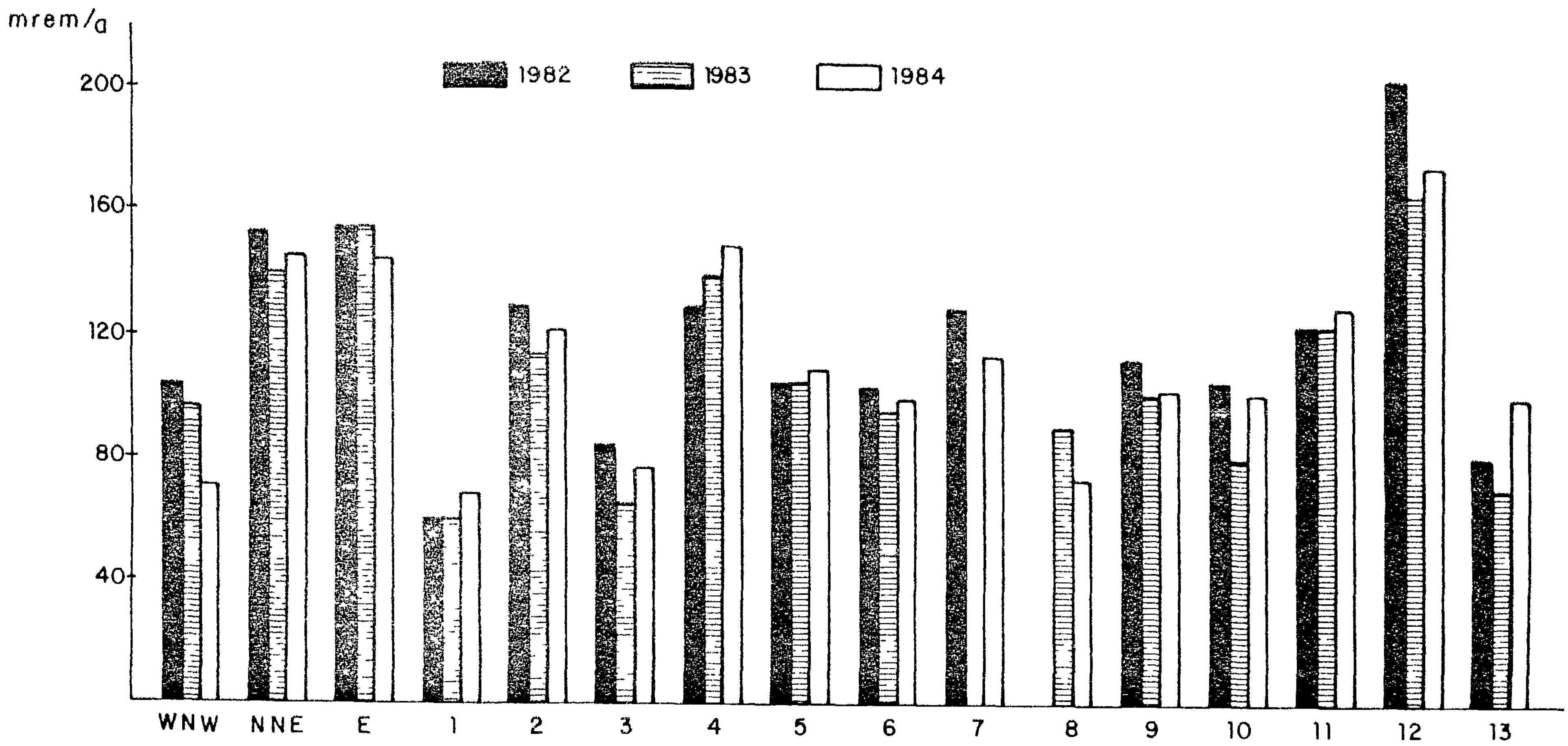


Fig 4 : Yearly LiF-TLD results for all measurement stations from 1982 to 1984 Stations are numbered according to the program description No data were available for station 7 in 1983 and for station 8 in 1982

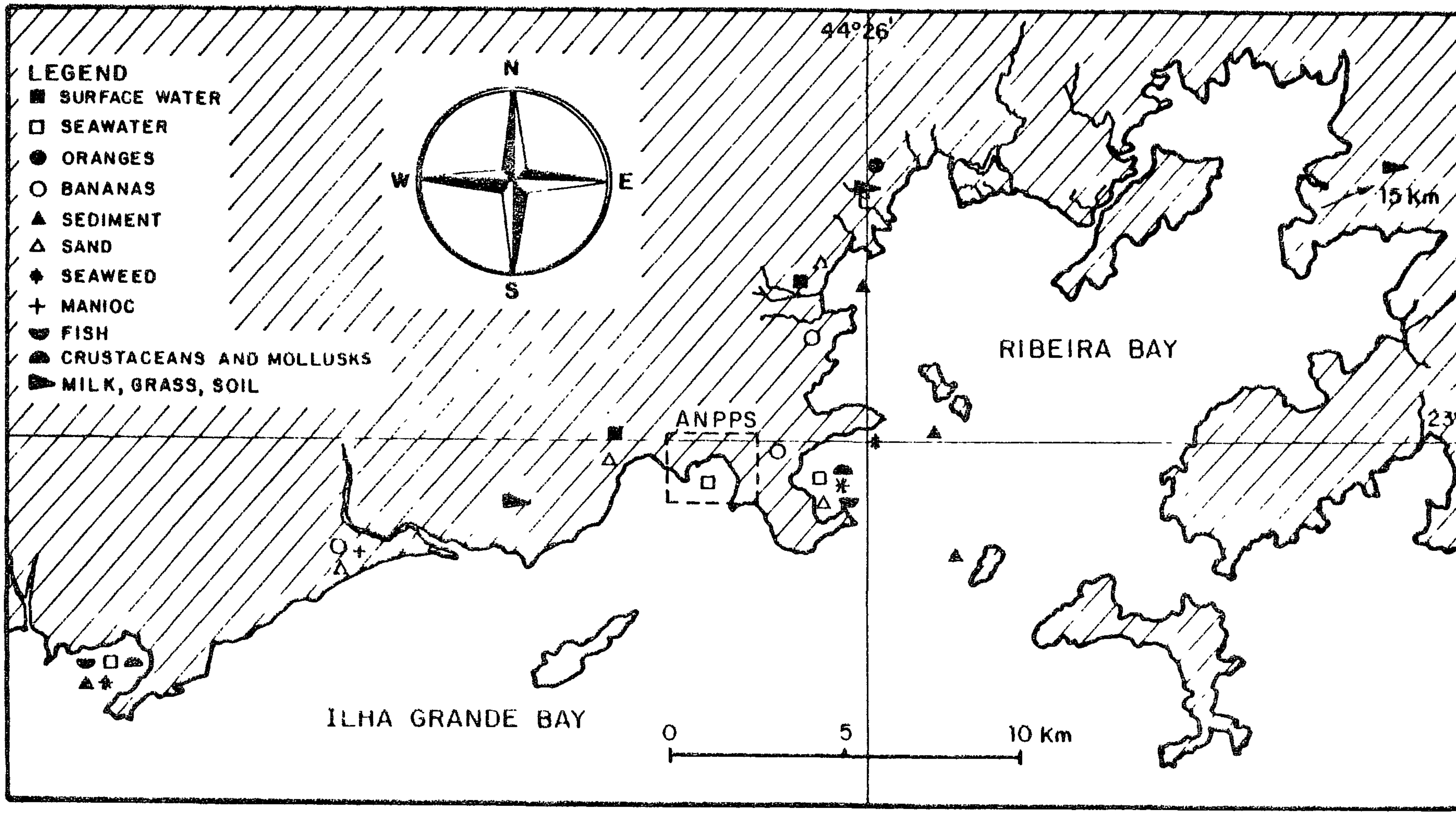


Fig. 5: Distribution of sampling locations around the Angra Nuclear Power Plant Site.

in sea water and surface water and of ^{131}I in air and milk.

Although the minimum detection limits for the measurement of fission and activation products by gamma spectrometry were continuously improved during the operational period, the only detectable artificial radionuclide was ^{137}Cs whose presence in milk, pasture, fish and manioc can be attributed to worldwide fallout. On the other hand, the direct radiation levels did not differ from those observed during the pre-operational period. Therefore, it is concluded that the operation of ANGRA NPP Unit 1 did not result in any radiological impact on the environment. The data presented come from studies during the monitoring period 1982-1984.

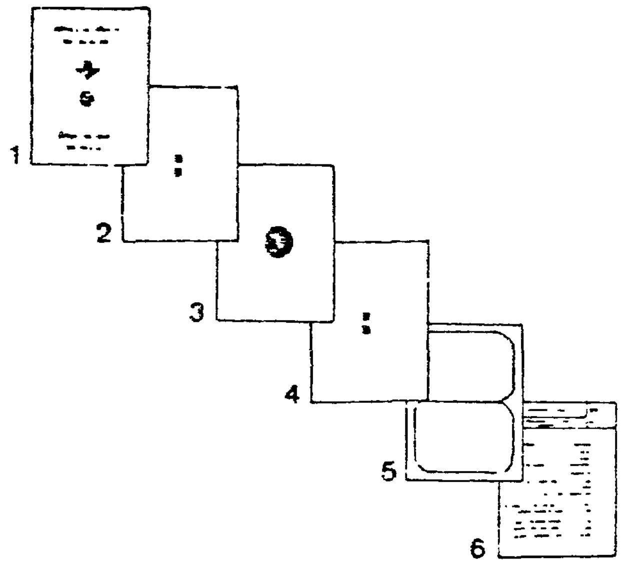
QUALITY CONTROL IN DENTAL RADIOLOGY

This will be presented as an example of IRD's efforts towards exposure reduction and calculation of population dose.

Since dental radiology and miniature photofluorography are responsible for 15 x 10^6 and 12 x 10^6 radiographs per year in Brazil respectively, they represent large sources of population exposure. Postal kits were developed for both techniques in order to enable large surveys to be carried out. The postal system for dental radiology which will be presented has the following objectives:

- to normalise skin entrance exposures for dental examinations;
- to access the quality of processing of intraoral radiographs in the dental clinics; and
- to collect data for evaluation of population dose in oral radiology.

Two postal kits have been designed, one to evaluate the quality of processing and another, described below, which evaluates field size, entrance exposure and half-value layer of the beam. The postal kit (fig. 6) is made up of six cardboard



1 - Front cover
2 - Two TLDs
3 - 3 mm Al filter
4 - Two TLDs
5 - Radiographic film
6 - Back cover

Fig. 6 - The postal dosimetry card for oral radiology

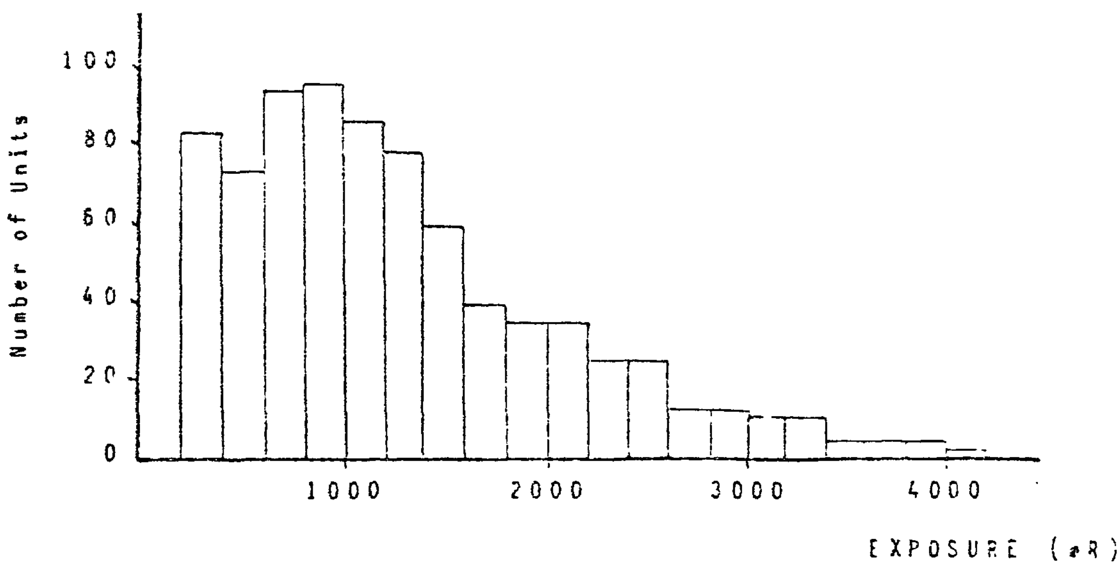


Fig. 7 - Skin entrance exposure distribution for upper molar region examination as determined with postal kit.

layers of 10 cm x 12 cm where are found: front cover, two TLDs, a 3 mm Al filter, another two TLDs, a radiographic film and a back cover. Dentists receive the card, fill in a questionnaire with technical details of equipment, irradiate the kit as if they were taking a radiograph of the upper molar region and return the kit to IRD, where the film is developed and the TLDs are read.

Results of a survey of nearly 800 X-ray tubes show that:

- entrance exposure varied from 200 mR to 4000 mR, the mean value being 1300 mR (fig. 7), and
- beam field size at entrance varied from 4.5 cm to 11 cm, when standard field size should be 6 cm.

This programme is helping dentists to identify and correct problems with their equipment and technique, consequently reducing exposure to patients.

RADIATION MEASUREMENTS DUE TO CHERNOBYL

a - Monitoring of planes and ships coming from the Northern hemisphere

Monitoring of the contamination levels of ships and airplanes coming from the Northern hemisphere was carried out from May 1986, making up a total of 25 international and 19 national flights. Smears and defrost water samples from the body of commercial planes were collected. For ships, smears of a 1 m^2 area were made of the external wall of ventilation towers, and cargo was examined.

The radioisotopes detected were: ^{95}Zr, ^{99}Mo, ^{103}Ru, ^{106}Ru, ^{131}I, ^{132}Te, ^{134}Cs, ^{137}Cs, ^{140}Ba, ^{141}Ce and ^{144}Ce. The highest activity measured was 525 Bq/l of ^{131}I in defrost water from planes.

Coal and sulphur brought as cargo aboard two ships which were at the port of Gdansk, Poland, at the time of the accident, were also analysed. Coal was contaminated with ^{103}Ru

(2.2 Bq/kg), ^{131}I (1.7 Bq/kg) and ^{137}Cs (0.9 Bq/kg) and sulphur was contaminated with ^{103}Ru (1.0 Bq/kg), ^{131}I (0.3 Bq/kg), ^{134}Cs (0.3 Bq/kg) and ^{137}Cs (0.5 Bq/kg).

So far, no contamination has been detected in the Southern hemisphere. Detection limits are compatible with those of usual radiochemical techniques for environmental control.

b - Individual monitoring

Brazilian citizens who were travelling through Eastern and Western Europe at and shortly after the time of the accident were monitored at IRD's whole body counter by one 8 in by 4 in NaI(Tl) detector (whole body chair geometry) and by one phoswich detector placed over the thyroid. The intake of some fission products (^{131}I, ^{132}I and ^{137}Cs) was observed in about sixty persons monitored until July 1986. Personal belongings were also examined. The analysis of urine samples confirmed the results obtained with the whole body counter. The individual effective dose equivalent due to such intake varied from 0 to 3.4 x 10^{-5} Sv (3.4 mrem). These results were presented last year at the 31st Annual Meeting of the Health Physics Society in USA.

c - Analysis of imported foodstuff

After the Chernobyl accident, the Ministry of Agriculture has requested that all imported foodstuffs have to be monitored and more than 600 food samples were analysed at IRD last year. The data presented in the table cover the period from May 1986 to March 1987. These data are now being used in a study of the impact of ingesting such foodstuff on population dose.

CONTROL OF IMPORTED FOODSTUFF (MAY 1986 - MARCH 1987)

FOODSTUFF (NO.LOADS)	COUNTRY OF ORIGIN	ACTIVITY 134 Cs (Bq/Kg)		ACTIVITY 137 Cs (Bq/Kg)	
		HIGHEST	LOWEST	HIGHEST	LOWEST
BEEF (96)	DENMARK, FRANCE, FRG, GREECE, HUNGARY, IRELAND, ITALY, NETHERLANDS, POLAND, UK, USA, YUGOSLAVIA	-	-	8.4 ± 0.8	< 0.4
BACON (3)	HUNGARY	-	< 0.5	-	< 0.7
PORKMEAT(5)	HUNGARY, SWEDEN	1.3 ± 0.6	< 0.7	3.0 ± 0.6	< 0.8
CODFISH(24)	NORWAY	1.5	< 0.8	8.1 ± 1.3	< 1.1
POWDERED MILK (247)	AUSTRIA, BELGIUM, DENMARK, CZECHOSLOVAKIA, FRANCE, FRG, IRELAND, NETHERLANDS, USA, N. ZEALAND	772 ± 79	< 0.9	1641 ± 79	< 0.9
CHEESE (21)	AUSTRIA, BELGIUM, FRG, NETHERLANDS, SWITZERLAND	13.1 ± 1.3	< 0.7	27.9 ± 2.0	< 0.8
WHEAT (1)	FRANCE	1.2 ± 0.5	-	1.7 ± 0.5	-

RESEARCH, EDUCATION AND TRAINING IN RADIATION PROTECTION IN BELGIUM

Rapporteur: R. Kirchmann, CEN/SCK, Mol

(Belgium)

Introduction

The general report (1) presented by Belgium pointed out that the development of nuclear energy for peaceful purposes in Belgium is accompanied by constant attention to keeping control over possible harmful effects in order to limit, as far as possible, adverse consequences for the health of the population and for the environment.

Protection of the population and workers against the hazards of ionizing radiation is also covered by general regulations fixed by Royal Decree (2). Verification of the application of the general regulations is the principal task of the Ionizing Radiation Protection Service (SPRI) established within the Public Hygiene Department of the Ministry of Public Health and the Family. It is also incumbent on the physical monitoring services of the facilities to ensure the application of the aforesaid regulatory provisions in facilities in classes I - III, depending on their nature. The physical monitoring service in such facilities is run by approved experts who fulfil the conditions set out in the general regulations.

Belgium relies heavily on nuclear power for the generation of electricity and this situation will remain unchanged in the coming decades. Since radiation protection in nuclear plants is one of the important aspects of personnel safety, it is logical that plant operators should focus attention on personnel training in this area.

Finally, the development of nuclear energy requires continuity in the effort not only in monitoring, but also in research, radiobiology and radioecology in particular, in order to arrive at a correct evaluation of the dose level resulting from exposure to ionizing radiation and to hand on the know-how gained.

I. TRAINING/EDUCATION

1.1 Universities

(a) There are radiation protection courses in the basic study programmes for medical students wishing to specialize. However, instruction in radiation protection is not given to some categories of medical practitioners and it is planned to extend existing radiation protection training to cover all those working in medicine and dentistry who use ionizing radiation and radioisotopes. A campaign to provide information about radiation protection for paramedical personnel and assistant staff who have not had university education is also recommended.

(b) Post-graduate courses

Some universities teach the knowledge necessary for the use of nuclear techniques and for expert training in medical and physical radiation protection. This report is not designed to give a list of such post-graduate courses. We shall merely note that the subjects taught include notions of atomic and nuclear physics and radiation, as well as the fundamentals of radiation protection, radiation pathology and radiobiology (health and radiation protection).

Furthermore, a post-graduate inter-university course in radiation protection as applied to the environment is being prepared.

1.2 Radiation protection training at nuclear power plants

1.2.1 Introduction

Radiation protection in nuclear power plants is an important aspect of personnel safety.

Personnel training is one of the topics that must be tackled because it is a special field which, apart from specialized studies, is not covered in schools.

1.2.2 Personnel categories

The personnel to be trained can be split up into two main groups:

- Personnel belonging to the company operating the plant;
- Personnel from outside enterprises.

(a) Personnel employed by the operator:

There are several different categories:

- Engineers in the "Monitoring" service responsible for radiation protection, among other things;
- Engineers in the "Production" services responsible for running the units;
- Engineers in the "Maintenance" services who deal with equipment maintenance;
- Radiation protection personnel ("Monitoring" services) responsible for daily monitoring;
- Personnel in the "Production" and "Maintenance" services who run and maintain the facilities respectively.

The radiation protection training of these various categories must be tailored to their specific responsibilities and occupations.

(b) Personnel from outside enterprises:

These people are involved either on a "permanent" basis (e.g. cleaning services) or during unit overhaul. They need to receive a modicum of information about radiation protection.

1.2.3 Radiation protection training

(a) Plant personnel:

Here is a brief outline of the types of training provided for each category of personnel.

- "Monitoring" engineers:

  These are civil engineers (university graduates). Unless their study or previous experience in the nuclear field is considerable, they must spend an additional year studying nuclear energy at one of the Belgian universities. This specialized study covers radiation protection in detail and the paper they write at the end of the course is generally concerned with this area.

  In addition, there are courses provided by Electricité de France (two weeks) or the plant itself (one week) to supplement their training in "power plant" applications.

  Finally, there are technical training courses to enable them to improve their training further on a practical level and/or to take refresher courses.

- "Production" engineers:

  Basic training is provided at the plant by the operator's specialized engineers. Courses provided by Electricité de France, identical to those given to "monitoring" engineers, are compulsory.

- "Maintenance" engineers:

  Training identical to that given to "production" engineers.

- Radiation protection personnel:

  In addition to basic training provided for all personnel (see following paragraph), there are more advanced internal courses and practical training for instructing such personnel and updating their knowledge.

- Personnel in other services:

  The basic training is provided by an approved body. There are regular refresher courses, of which a particular feature is time spent on instruction sites.

(b) Personnel from outside enterprises:

The persons concerned are given training by their employers. However, a video system explaining the special features of the Tihange site (entry to and exit from the controlled zone, dressing/undressing, irradiation/contamination, special sites, dose meters, posting of signs, alarm signals, etc.) is shown to all employees - in small groups - followed by questions and answers. This takes place before they are allowed access to the facilities.

1.3 Basic courses and further training

The taking on of trainees by research laboratories and enterprises, together with the organization of specialized training periods (physical monitoring, radiobiology, radioecology, nuclear metrology, etc.) offers another way of furthering the development of radiation protection techniques and the transfer of knowledge.

By way of example, during the period 1985-1986, 156 trainees from 34 countries worked at the CEN/SCK laboratories and, of these, 22 trainees from 12 countries took part in work concerned with radiation protection.

II. RESEARCH

The research programme carried out in Belgium in the field of radiation protection includes biological projects (somatic and genetic effects of ionizing radiation) and programmes concerning the behaviour of radionuclides in the environment (dispersion, migration, transfer), as well as the development of nuclear metrology and dosimetry. It should be noted that research into radiobiology and radioecology needs to be backed up by special arrangements, such as the breeding of laboratory animals and larger species (experimental farm), as well as by the cultivation of plants under glass and in test plots.

With regard to the worrying problem of continuity in the field of radiation protection and, more particularly, radiobiology and radioecology, the continuation of certain high-level research activities makes it possible to promote and test the potential of young researchers.

2.1 Transfer and dispersion - ecosystems

The purpose of these activities is to maintain the knowledge gained in the area of atmospheric dispersion, particle pathways and dose evaluation. This objective is a sine qua non for rapid response in the event of an accident and the ability to assess the situation. Various studies are in progress: they are principally covered by contracts with the Commission of the European Communities and are therefore co-ordinated and integrated throughout Europe.

Research into the metabolism of radionuclides present in wastes from nuclear facilities deals with the biological behaviour of technetium and the transuranic elements and the transfer of tritium (in elemental and organic form) and carbon-14 to mammals. This research receives financial support from the Commission of the European Communities.

2.2 Somatic effects of ionizing radiation

This heading includes research into stochastic and the non-stochastic effects (occurring above a certain threshold).

Research in stochastic effects covers three aspects: induction of leukaemia and osteosarcoma by radiation; effects of exposure to gamma rays and neutrons on the induction of liver cancers; influence of age on the induction of cancer by radiation, whether or not it is combined with a chemical carcinogen.

Research into non-stochastic effects involves comparing damage done by internal exposure to radiation on the haematopoietic and stromal systems; delayed effects of pre-natal exposure to radiation on the central nervous system; early and late effects of perinatal or adult exposure to radiation on the haematopoietic and immunological systems; effects of radiation on plants.

This research is covered by contracts with the Commission of the European Communities - and therefore forms part of a co-ordinated and integrated system - or by a contract with the Scientific Medical Research Fund (FRSM).

2.3 Genetic effects of ionizing radiation

The aim of research into the genetic effects of ionizing radiation is to help improve knowledge in various fields: evaluation of the incidence of chromosome aberration induced in lymphocytes in the bloodstream following exposure to low doses of X-rays; radiation-induced structural aberration of the chromosomes in the somatic cells of mammals, and morphological and cytogenetic studies of embryonic sensitivity to low radiation doses.

This research receives the financial support of the Commission of the European Communities.

2.4 Radiation exposure - dosimetry

The aim of the studies and research is to evaluate the dose resulting from exposure of the Belgium population to radiation of natural and/or artificial origin. There is a specific programme of radiological monitoring around nuclear facilities, under a contract between the Ministry of Public Health (Ionizing Radiation Protection Service - SPRI), and the CEN/SCK-IRE.

Conclusions

A substantial effort is being made in the field of radiation protection, which is justified by the size of the Belgian nuclear programme.

The universities have an important part to play, not only in the field of education, but also in informing the public. It should be stressed that, if education is to be useful, it cannot be dissociated from research. Research can also be useful for the subject of radiation protection by encouraging young researchers and thus ensuring the necessary continuity of interest.

References

1. "The Belgian Nuclear Industry and Associated Research and Development Programmes", general report.

2. Ministry of Employment and Labour and Ministry of Public Health and the Family - 28 February 1963 - Royal Decree governing general regulations for the protection of the population and workers against the hazards of ionizing radiation. Moniteur Belge, 16 May 1953, 5206-5291.

3. Official Journal of the European Communities - Directive 84/466/Euratom. Council Directive of 3 September 1984 amending Directive 80/836/Euratom with regard to basic standards for the health protection of the population and workers against the hazards resulting from ionizing radiation, L265, 5 October 1984.

4. J. L. Garsou - Radiation Protection Training of Medical and Paramedical Personnel: the current situation and proposals. Bulletin of the Belgian Radiation Protection Association, Vol. 11, No. 1 (1986), 19-30.

5. L. de Thibault de Boesinghe - The Importance of Radiation Protection in the Training of Doctors and Paramedical Personnel. Bulletin of the Belgian Radiation Protection Association, Vol. 11, No. 1 (1986), 31-40.

List of abbreviations

CCE Commission des Communautés Européennes
(Commission of the European Communities)

CEN/SCK Centre d'Etude de l'Energie Nucléaire
(Centre for Nuclear Energy Research)

FRSM Fonds de Recherche Scientifique Médicale
(Scientific Medical Research Fund)

IRE Institut National des Radioéléments, Fleurus
(National Radioisotope Institute, Fleurus)

IRSN	Institut Royal des Sciences Naturelles (Royal Natural Sciences Institute)
KUL	Katholieke Universiteit Leuven (Louvain Catholic University)
RUG	Rijksuniversiteit Gent (Ghent Royal University)
SPRI	Service de Protection contre les Radiations Ionisantes, Bruxelles (Ionizing Radiation Protection Service, Brussels)
UCL	Université Catholique de Louvain (Louvain Catholic University)
UEL	Université de l'Etat de Liège (Liege State University)
ULB	Université Libre de Bruxelles (Brussels Free University)

Annex 1

GENERAL TRAINING PROGRAMME FOR PERSONNEL WORKING IN RADIATION PROTECTION AT TIHANGE NUCLEAR POWER PLANT

1. Outline of the radioactive risk at the plant.
2. Purpose of radiation protection and attitude of workers to the risks.
3. Radioactivity, decay, half-life.
4. Nature of radiation.
5. Effect on matter: ionization, absorption.
6. Concepts of dose and dose rate.
7. Mechanism underlying irradiation.
8. Surface and atmospheric contamination.
9. Effects of radiation on man, standards for protection.
10. Risk evaluation.
11. Measurement and interpretation.
12. Protection equipment.

N.B. These courses are given by a specialized engineer from an approved body.

Annex 2

SUPPLEMENTARY TRAINING PROGRAMME FOR PERSONNEL WORKING IN RADIATION PROTECTION AT THE TIHANGE NUCLEAR POWER PLANT

1. Basics of mathematics.
2. Basics of nuclear physics.
3. General principles of radiation detection.
4. Radiation detection.
5. Various types of detectors.
6. Ionization chambers, proportional counters.
7. Geiger-Müller counters for use during exposure to radiation and contamination.
8. Alpha, beta and gamma measuring techniques.
9. Radiobiology.
10. Shielding.
11. Dose units and dose rates.
12. Design of the controlled zone.
13. Origin of radioactive products.
14. Controlled zone exits.
15. Gamma radiography at the plant.
16. Site marking.
17. Performance of radiation protection instruments and practical exercises.
18. Body decontamination, action in the event of an accident (three hours).

N.B. These courses are given by engineers and experts in the station (specialized personnel) with the support of the medical service (doctor and nurses) and outside instructors (principally design consultants).

Annex 3

SUPPLEMENTARY TRAINING PROGRAMME FOR ENGINEERS AT THE TIHANGE NUCLEAR POWER PLANT

1. Fundamentals of nuclear physics. Basic elements of radioactivity.
2. Origin of irradiation and contamination sources in a nuclear power plant.
3. Biological effects of ionizing radiation.
4. Interaction of radiation with matter. Introduction to the theory of radioactivity measurement.
5. Radioactivity measurement: principles and techniques, site instruments, practical work.
6. Radioactivity measurement: fixed system.
7. Personnel dosimetry: external dosimetry, whole body gamma radiography.
8. Legislation and standards for radiation protection.
9. Radioactive effluents: exploitation, legal provisions.
10. Prevention and policy in radiation protection: ALARA principle, special materials and tools.
11. Role of industrial medicine in a nuclear power plant.
12. Transport of radioactive materials.
13. Radiation protection instructions. General principles underlying individual and collective protection.
14. Reaction in the event of a radiation accident affecting a person.
15. Body decontamination techniques. Inspection of a decontamination area.
16. Emergency plan in the event of a radiation accident. Simulated exercises with accidental releases and calculations.

N.B. These courses were given by:

- Engineers from the plant specializing in the areas involved;
- Doctors from the medical service;
- An EDF medical specialist;
- Engineers from an approved body;
- Representative of the Ministry of Public Health.

"ARGOS": A COMPUTER TOOL FOR RAPID DECISION-MAKING IN CASE OF NUCLEAR EMERGENCIES

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(Denmark)

In the event of a nuclear emergency, time is one of the most crucial factors.

The emergency preparedness experts need time to evaluate the often scarce information available at the beginning of the emergency. The monitoring teams need time for taking proper measurements of radioactivity and radiation levels if any - a zero reading is also of importance.

Time is needed for the transfer of those readings from the monitoring teams to the technical evaluation centre. And time is here needed for transforming the readings to an understandable background for evaluation.

Further, time is needed by the decision-makers to choose which measures are to be decided upon:

(i) The public must be informed as soon as ever possible - but about what?

(ii) Any protection measure such as: an order to shelter the population in a certain region, or an order to evacuate all people in a certain region, has to be based on the best amount of information.

One fact is obvious: time is of the utmost importance in all these complicated areas having enormous, both national and international, consequences.

We have, in Denmark, developed a computer tool which we find useful especially when the concern is:

(i) saving time; and

(ii) presenting observations;

in the most easily understandable way for the decision-makers.

Our base for developing the ARGOS-system was the Danish emergency planning for the Swedish Barsebäck Nuclear Power Plant, 20 km from the coastline of Copenhagen, the capital of Denmark. Two boiling water reactors, each of 600 megawatts of electricity, are situated north of the Swedish city of Malmö. In the event of an alarm, many monitoring teams are immediately despatched both in Denmark and in Sweden.

From Denmark, a helicopter is also immediately despatched for measurements at distances upto 5 km from the plant. Permission for this flight is immediately sought from the Swedish authorities.

In order to save time, we have developed a computer system which eliminates many of the sources of errors and misunderstandings in a communication system which otherwise consists of:

(i) writing down measurements;

(ii) oral message transfer by telephone or radio;

(iii) writing down the message;

(iv) eventually another transfer, orally and written;

(v) transfer by hand to wallmaps of figures and plotting of curves;

(vi) evaluation on the basis of hand-drawn sketches.

We cannot yet eliminate (i) and (ii), but it will be possible for us to do so in the very near future.

We have now eliminated (iii), (iv) and (v) by means of the introduction of four terminals connected to the computer system for direct transfer of data from four places. (vi) can now be made on the basis of iso-intensity curves drawn on a relevant map by the computer.

Two identical computers are placed: (i) at the Copenhagen Emergency Co-ordination Centre; and (ii) at the Risø Technical Evaluation Centre. One terminal is placed in Hillerød and another with the Swedish authorities in the city of Malmö. Isodose pictures can be combined with the population distribution and the system can compute the collective dose to be expected in the exposed area at chosen times.

Now, after the Chernobyl accident, it has been decided that a nationwide Nuclear Emergency Preparedness Plan be put into effect by 1 January 1988.

Although Denmark has no nuclear power plants itself, it is surrounded by several plants in neighbouring countries: Sweden, German Democratic Republic and the Federal Republic of Germany.

We are now in the process of creating ten permanent early warning stations with the task of detecting even small releases of man-made radioactivity. These will be connected to an on-line computer which will be installed at the Research Centre, Risø. In the event of an alert situation - which will only be declared after evaluation of the situation by qualified experts - the system will communicate into the ARGOS-system which will then present the nationwide situation for evaluation. International communication of the situation to other authorities will then be possible from the ARGOS system.

THE SAFETY OF WATER REACTORS: THE FRENCH APPROACH

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(France)

This report provides an outline of the French "philosophy" of safety and the resulting approach to the technical analysis of problems related to the safety of pressurized water reactors (PWRs).

It should be recalled that the French nuclear power programme is based on the design, construction and operation of series of standardized units, the only adjustments being those due to the sites selected for the various units of the same series. After the two Fessenheim units and the four Bugey units, the experience gained has made it possible to plan, in succession, the series of 900 MW(e) units (CP1 followed by CP2), the series of 1,300 MW(e) units (P4 followed by P'4), and the series of 1,400 MW(e) units (N4).

The first units were modelled on American facilities under construction (Beaver Valley for Fessenheim, North Anna for Bugey). At that stage, EDC (Electricité de France) and the authorities responsible for safety relied extensively, with modifications to suit French conditions, on the American regulations (10 CFR50 and regulatory guides) in matters relating to the safety or safety analysis of the French units.

This initial approach was gradually broadened further by the development of probabilistic approaches and by taking into consideration serious accidents. The various aspects which will be described in greater detail below have greatly benefited from the standardization of the units, which ensures, in particular, that feedback from experience gained can be extensively used in improving their safety.

* * *

Accident prevention is based primarily on a deterministic approach, the aim of which is to prove that, in the various situations regarded as foreseeable (normal operation, incident situations and accident situations), adequate containment of radioactive materials is assured. This containment is achieved through "barriers", and the situations concerned arise from the application of a concept of "defence in depth".

Maintenance of the containment of the radioactive products is based on the interposition of "barriers" between those products and workers or members of the public. In the pressurized water reactors built in France, a systematic distinction is drawn between three basic "barriers": the cans containing the fuel, the shell of the primary pressurized circuit and the containment of the nuclear steam supply system. The behaviour of the "barriers" is examined under all conditions corresponding to the normal operation of the plant and to the incident and accident situations regarded as foreseeable. A release of radioactive products can only occur in the event of a loss of leaktightness affecting all the barriers.

The concept of "defence in depth" used in defining situations regarded as foreseeable covers:

Planning, construction and operation of facilities to ensure that the plant is inherently resistant;

Installation of control or protection systems capable of returning the plant to its normal mode of operation in any foreseeable cases of transients and incidents;

Due provision for accidents which might occur despite the preventive measures adopted under the two previous points, and planning of protection systems capable of limiting the consequences of such accidents.

As far as the application of this concept is concerned, the following points should be noted.

(1) The fact of designing a component or a system for a given situation does not rule out the possibility that it will break down in that situation. If the consequences of such a breakdown are considered unacceptable, additional measures must be taken to limit or prevent them. Accordingly, every precaution is taken to ensure that the pressurized circuits can withstand the maximum stresses to which they may be subjected; at the same time, their rupture is taken into account in accident research. Exceptions to this rule can only be made if the preventive aspect is sufficiently "reinforced": the sudden rupture of the vessel in pressurized water reactors is thus ruled out, by virtue of the precautions taken in design, manufacture and inspection throughout the life of the plant, so that defects which might lead to more serious breakdowns can be detected in time (this point is also subject to specific regulations which are particularly closely scrutinized by the authorities).

(2) Since it is not possible to examine all accident situations regarded as foreseeable, the operator and safety authorities have agreed to examine a limited list of accident situations drawn up so that they are representative of the hazards involved; each situation is selected and studied in such a way that its consequences are greater than those of the comparable events which it is supposed to represent (the "accident envelope" approach).

(3) The failures which might call into question both the measures taken to prevent accidents and the means for limiting their consequences must be identified; measures have been adopted to ensure that failure does not give rise to unacceptable consequences. Accordingly, the total breakdown of the external and internal electricity supply is liable to cause the loss of primary coolant (leakage from the joints of the primary pumps), while the protection systems (without benefit of current) would be unable to operate in such conditions. Similarly, a fire would act as a cause of "common mode" failure.

The problem which then arises is of knowing "how far to go" and of ascertaining what accident situations should be borne in mind when designing the plant. Generally speaking, as mentioned above, the deterministic approach allows for a conventional list of situations classified according to frequency and the consequences of those situations must, in each category, still be lower than values which are all the higher when the situations are less likely.

At this stage, it should be emphasized that French regulations require, for each site, licences for the discharge of gaseous and liquid radioactive wastes; these licences specify, on a case-by-case basis, the maximum overall activity for which the discharge is authorized, with specific activity limits for certain radioactive substances. On the other hand, French regulations do not specify limits for dose equivalents likely to be delivered to members of the public in accident situations; the radiation consequences of such situations (conventional operating conditions and situations arising from outside events) are calculated without reference to limit dose equivalents, but are submitted and regarded as acceptable in the course of specific licensing procedures at each plant in which the Ministry responsible for public health plays a role.

For the design of its facilities EDF has fixed the following limits, which have been accepted by the safety authorities:

Frequency category	Estimated frequency (per year)	Maximum consequences in in terms of radiation
1	1	Limited by licences for radioactive effluent discharges
2	$20^{-2} - 1$	
3	$10^{-4} - 10^{-2}$	0.5 rem (whole body) 1.5 rem (thyroid)
4	$10^{-6} - 10^{-4}$	15 rem (whole body) 45 rem (thyroid)

The annex contains the conventional list of operating conditions selected for the 1300 MW(e) units.

* * *

The original deterministic approach has been added to by the development of probabilistic approaches. Historically speaking, an approach of this kind was first introduced to deal with the measures adopted with regard to off-site events in order to bring the situations arising out of the design events selected nearer to the conventional list of on-site accidental situations.

To give an illustration - it was on the basis of study of the probability of aircraft crashing onto PWR nuclear power plants that the French safety authorities accepted the fact that different series of units could be designed in a manner that allowed for the idea of only a crash by "general" aircraft (taking into consideration a "hard" projectile represented by a CESSNA 210 plane weighing 1.5 tons and a "soft" projectile represented by a LEAR JET 23 weighing 5.7 tons), and excluded military and commercial aircraft; this led to the rejection of certain sites.

Nevertheless, in 1977 when the major technical options for the 1300 MW(e) units were being studied, the French safety authorities fixed an overall probabilistic target in those terms:

"Generally speaking, the design of facilities for a unit comprising a pressurized water reactor should be such that the overall probability of the unit giving rise to unacceptable consequences should not exceed 10^{-6} per year.

"Consequently, when a probabilistic approach is used to estimate whether a set of events should be taken into account for the design of such a unit, it has to be considered that those events should in fact be considered if the probability that they might lead to unacceptable consequences is more than 10^{-7} per year, with this value not being exceeded in the case of the given set of events unless it is possible to prove that the probability calculations made are rather pessimistic.

"Moreover, it appears necessary for Electricité de France to continue its efforts aimed at the use of probabilistic approaches for the greatest possible number of events as soon as possible.

"In applying the aforesaid Electricité de France should in each individual case study whether the simultaneous failure of redundancies in safety-related systems should be taken into account in the design of units comprising a pressurized water reactor ... For such studies use could be made of realistic hypotheses and calculation methods".

This text calls for several comments:

(1) The overall target is fixed in terms of "unacceptable consequences"; accordingly, it is stated above that these "unacceptable consequences" are not defined by any legislative or regulatory document - they must in fact be assessed in political terms, with allowance, if necessary, for the effects associated with sites and the possibilities for action to protect the population.

(2) The probability of 10^{-6} per year is a "target" value for a unit and EDF is not required to prove that the target has been reached; the "target" has nevertheless been considered reasonable, bearing in mind the results published in the WASH 1400 report and the improvements made to the design of French units compared with the pressurized water reactor plant studied in the report. Justification of the design features selected to avoid any unacceptable hazard is still, at the outset, based on deterministic studies and not on an overall probabilistic analysis.

In a letter containing comments addressed to EDF in 1978 on this subject, the safety authorities clearly stipulated the framework for probabilistic studies required from EDF:

"I would like to stress ... that my wish to ensure the use of a probabilistic approach for the greatest possible number of sets of events does not involve the direct use of these methods for design of units containing a pressurized water reactor. Probabilistic verifications can be easily carried out a posteriori to show the soundness of the measures foreseen and such studies can, furthermore, improve the definition of the deterministic criteria used, should such be necessary, for designing later units.

"The terms of my letter ... (of 1977) ... do not mean, either, that the safety of a pressurized water reactor unit should now be proved by an exhaustive probabilistic analysis. On the other hand, the use of probabilistic approaches should make for better justification, i.e. improvement, of the definition and classification of situations considered when designing a unit of that kind."

(3) The value 10^{-7} per year is used more directly in the operational sense and the approach referred to above for off-site events applies this value when considering, for example, several sets of events for air crashes: the probability of a "general" aircraft crashing on a power plant is in France such that pressurized water reactor plants adopt systematic protection measures against a crash of that kind, wherever the site; on the other hand, the probability of a

commercial aircrash on a power plant is rather low in France, excluding the areas around airports, so that there is no need to adopt such measures in that regard; as far as military aircraft are concerned, the matter is examined for each site proposed so as to be sure that the site is really acceptable. The value 10^{-7} per year also makes it possible to deal with the problems of combinations of off-site events and conventional operating conditions.

It should be pointed out, however, that the value 10^{-7} per year is not considered a "cut-off" value above which design features must definitely be adopted. How to tell whether such features should be in fact adopted is considered in each individual case by means of a critical study of the assumptions made and by bearing two important aspects in mind:

(a) The overall target for risk: for example, in order to stay within the area of outside events it is possible to accept a higher degree of vulnerability with regard, say, to air crashes in view of a lessser degree of vulnerability to explosions; one has to bear in mind the number of sets of events for which the probability of their leading to unacceptable consequences is greater than 10^{-7} per year;

(b) The cost of the measures to be implemented as against the benefit expected from the safety standpoint;

(4) As opposed to the conventional deterministic approach, for which studies are made on the basis of the pessimistic assumptions and calculations, the probabilistic approach, if it is to be fully successful and permit greater consistency in the measures adopted to prevent any unacceptable risk, is based on values as realistic as possible both for assessing probabilities and for evaluating consequences.

The use of probabilistic approaches has enabled us to show the need for additional measures aimed at a satisfactory level of safety in certain situations that do not figure in the conventional list of operating conditions. The letter of 1977 quoted above requested EDF to give particular attention to the probabilities and consequences of:

(a) Failure of the scram system during transients calling for the actuation of that system;

(b) Failure of one of the systems for removing heat produced by the reactor to the "cold source" or from the "cold source";

(c) Simultaneous failure of the entire power supply.

As a result of these studies additional measures were in fact put into effect on the different series of units, consideration being given to the state of construction of them. This point can be illustrated by the case of simultaneous failure of the power systems.

Electrical power is supplied to pressurized water reactor plants by four independent sources: two outside sources based on the EDF distribution grid, and two inside sources each consisting of a generator working on diesel oil; there is in addition a possibility of switching over the turbo-generators so that they power their own auxiliary systems.

These power sources feed the two independent power distribution lines (line A and line B) for the safety and protection equipment by means of two control panels known as "back-up panels" (6.6kV LHA and LHB); one generator unit is assigned to line A and the other to line B.

When the studies mentioned above were made it became clear that in the case of the 900 MW(e) units simultaneous failure of all the electric power systems could rapidly lead to serious consequences. This kind of failure involves breakdown of the injection of water into the primary pump joints, which means loss of integrity in the primary circuit within a period of about three hours. The corresponding probability is of the order of 2 x 10^{-5} per year, taking into account the various initial states possible in the unit: half of this probability stems from failure of the power sources in the true sense, and the other half comes from the failure of the two control panels LHA and LHB.

This situation led EDF to examine various additional measures by which to reduce the probability of a serious accident. In this way, use of a gas turbine installed at the site and the implementation of methods for re-energizing a 6.6 kV control panel by means of a generator belonging to a neighbouring unit make it possible to reduce considerably the probability of a serious accident resulting from the simultaneous failure of power sources (to about 10^{-7} per year). The probability of a serious accident resulting from the simultaneous failure of all the power systems is, however, reduced only by a factor of about 2, since the repair of the back-up panels assumed to have failed cannot normally be made quickly.

To increase the time available to the operator in such a case one has to ensure the injection of water into the primary pump joints so as to avoid the loss of integrity in the primary circuit until the shutdown stage where the injection is no longer necessary. It was found that this could be done by an existing pump (test pump) provided it was coupled to a turbo-generator unit supplied with steam from the steam generators.

The set of devices required to put into effect the measures described above is being introduced for the 900 MW(e) units (procedure H3); in this way the probability of a serious accident is not more than 5 x 10^{-6} per year, the dominant sequence corresponding to a simultaneous failure of the back-up control panels lasting longer than the water-holding capacity of the steam generator emergency feed system, the water serving to remove the residual heat.

To reduce the probability of a serious accident still more, all that is needed is to have a way of supplying water to the steam generators. This supply should be possible by gravity feeding of the emergency steam generator supply tank and use of a condensor extraction pump. Serious accident probability is then reduced to about 10^{-7} per year.

One can see from this example how reflections based on a probabilistic approach have enabled us to reduce the risk associated with simultaneous failure of the power supplies of a 900 MW(e) unit to an acceptable value without fundamentally changing the initial design based on a conventional list of operating conditions.

Obviously, the data from these studies were borne in mind during the designing of the 1300 MW(e) and N4 units without any need to amend the conventional list of operational conditions. This is why, for these standardized units, the set of measures described above (including gravity backfeeding of water to the emergency steam generator feed tank) has been incorporated into the design of the corresponding units and the probability of a serious accident resulting in the case of a 1300 MW(e) unit from the simultaneous failure of all the power feed systems has therefore not risen above about 10^{-7} per year since their commissioning; for the N4 series some new improvements are now being studied.

* * *

Speaking more generally, the studies on probabilities and consequences of the total failure of redundant safety-related systems have shown that in order to attain the overall safety goal it has been necessary to adopt measures in addition to the automatic systems already featured in the initial design.

For example, special operational procedures have been worked out and defined that are known as H procedures (H standing for "beyond the design basis", but it would be more correct to say "at the design basis limit"). The H3 procedure has been described above, but it should be made clear here that there are in all five H procedures:

H1: loss of outside cold source;

H2: total failure of systems supplying water to the steam generators (normal and emergency supply);

H3: total failure of power sources (outside and inside);

H4: reciprocal emergency action by the systems for spraying the containment, and emergency low pressure injection during the recycling phase;

H5: protection of river bank sites against flooding that exceeds the reference flood (thousand year recurrence).

A supplementary procedure - U1 - is designed to improve still more the prevention of core melt accidents by an approach involving the state of the cooling system for the steam supply during an accident when there is no idea of the sequence of events that lead up to that situation.

However, despite all the precautions mentioned above for avoiding core degradation we cannot absolutely rule out the possibility of serious accidents involving core melt and partial or substantial loss, after some delay, of the containment of radioactive material in the vessel. In this respect, for a given scenario we can almost always imagine another scenario which would be worse by assuming an additional failure; it is quite clear that as one progressively considers more and more serious scenarios the probability that they may occur tends to zero. So where should one stop? Is it necessary, for purposes of protecting the population and adopting measures to that effect, to fix a new probability threshold and determine the maximum radioactivity releases corresponding to it?

Two points of criticism can be made with regard to this approach:

(1) The more improbable the events are, the greater the uncertainty of the calculation of their probability, so that the calculation itself does not have much significance;

(2) Above all, calculation of the source terms during such accidents circumvents the major problem of studying the control of the course of these accidents by a series of appropriate actions.

In the French approach to serious accidents no attempt is made to classify the serious accidents in any precise manner and use is made of the expression "source term" in a restricted sense: a "source term" is a typical release characteristic of a class of accidents; it is taken into consideration in order to define the action to be foreseen with regard to this class of accident, for the purpose of the ultimate protection of the population, as part of the preparation of emergency plans (on-site emergency plan for the power plant and special plan of action off-site).

As a result of studies carried out with French-designed water reactors, three reference source terms have been identified corresponding to three major categories of accidents, all of which cover a complete core melt-down. They are in order of descending seriousness:

- Source term S_1 for accidents involving early rupture of the containment (several hours after the accident starts): a typical example of such accidents is the α mode according to the terminology of the WASH 1400 report;

- Source term S_2 for accidents leading to releases outside the containment directly into the atmosphere as a result of a delayed loss of leaktightness, after a period of one or several days (example: mode δ);

- Source term S_3 for accidents leading to indirect releases because of the existence of pathways of transfer with retention between the containment and the outside air (example: mode ε).

The corresponding release levels have been evaluated on the basis of information contained in the WASH 1400 report. For these three source terms, they are, respectively, a few tens of per cent, several per cent and several per mille of the fission product content of the core for volatile products, apart from the rare gases released almost entirely in all three cases.

Source terms and fractions of activity released from core

Type of containment failure		S_1	S_2	S_3
		Early	Delayed	Late
	Rare gases	80 %	75 %	75 %
I	Organic	0.6%	0.55%	0.55%
I	Non-organic	60 %	2.7 %	0.30%
	Cs	40 %	5.5 %	0.35%
	Sr	5 %	0.6 %	0.04%

With regard to accidents entailing source term S_1, the French approach seeks to show that for French reactors with large containment systems these accidents can be ruled out either for physical reasons (impossibility of describing a train of events on a realistic basis) or else because they are too improbable: this relates to α and γ modes for rupture of the containment following on a steam explosion and a hydrogen explosion, respectively.

Furthermore, special emergency action plans of the public authorities have been worked out in France after consideration of reasonable chances for evacuation and containment of the population. This idea had led to envisaging the possibility of the following action: potential evacuation of the population to a distance of 5 km and containment of other persons up to 10 km within a period of 12-24 hours following the start of the accident.

Comparison of the scope of these measures with the assumed level of the radioactive releases indicates that they are compatible if the releases do not exceed the characteristics of source term S_2.

It now remained to deal with the case of accidents that could lead to releases through the above-ground part of the containment after a delay of the order of, or greater than, a day: this applies particularly to a pressure increase above the design value in the containment (mode δ), or serious leakage from the latter (mode β). For this purpose measures have been studied for improving the last containment barrier; these are emergency procedures, known as U, by means of which, using simple devices, the consequences of accidents can be limited irrespective of their cause.

To be more exact, procedure U2 stipulates the action to be taken if a fault in the insulation of the containment is detected during an accident.

Procedure U4 aims at avoiding any direct release of radioactive products through the drainage devices placed inside the concrete raft under the vessel well, and procedure U5 permits monitored and filtered releases through a special filtration system with a gain on the aerosol releases of the order of a factor of 10 (levelling of the containment pressure). Hence the source term S_2 can be reduced to the level of the source term S_3.

The set of measures described above enables us to deal with the question of serious accidents from the standpoint of the acceptability of their consequences. Since the Chernobyl accident thinking on the topic of managing an accident and the post-accident situation has continued.

* * *

The set of actions described above is aimed at improving, from a conventional deterministic approach, the consistency of all measures adopted to ensure the safety of facilities. Overall probabilistic evaluations are under way so that we can better assess the strengths and weaknesses, if any, of the 900 MW(e) and 1300 MW(e) units.

In any event, monitoring remains imperative in order to detect any deviation of the safety level of facilities. A special effort, which is not dealt with explicitly in this paper, is thus devoted to studying feed-back in order to derive from it valid information, more especially, to detect events foreshadowing serious accidents.

Annex

CONVENTIONAL LIST OF OPERATIONAL CONDITIONS SELECTED FOR 1300 MW(e) UNITS

- Average frequency incidents the consequences of which should remain extremely limited:

Uncontrolled withdrawal of the control rods of a subcritical reactor

Uncontrolled removal of reactor control rod group operating at power

Wrong positioning, fall of a rod or group of rods

Uncontrolled boric acid dilution

Partial loss of primary flow

Start-up of an inactive loop

Total loss of load - turbine trip

Loss of normal feedwater

Wrong functioning of normal feedwater

Loss of external power supply

Excessive load increase

Opening of a pressurizer valve at the wrong time (momentary depressurization of primary circuit)

Opening of a secondary circuit valve at the wrong time

Start-up at the wrong time of a safety injection or boration.

- Very infrequent accidents the consequences of which should be fairly limited:

Loss of primary coolant (small ruptures)

Opening at the wrong time of a pressurizer valve (prolonged depressurization of the primary circuit)

Slight rupture of the secondary-circuit piping

Total loss of primary flow

Wrong positioning of a fuel assembly in the reactor

Removal of a control rod group at full power

Rupture of the chemical and volume monitoring circuit tank

Rupture of the storage tank of the gaseous effluent treatment circuit.

- Serious and hypothetical accidents the consequences of which should still be acceptable:

  Fuel handling accident

  Sizable rupture of the secondary circuit (steam or water)

  Blocked rotor of a primary motor pump

  Ejection of a regulating rod group

  Feasible loss of primary coolant

  Complete rupture of a steam-generator pipe.

ONTARIO HYDRO'S SYSTEMS APPROACH TO RADIOACTIVE MATERIALS MANAGEMENT

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INTRODUCTION

Ontario Hydro is a large provincial utility serving Canada's most heavily industrialized province, Ontario. In a system of 80 hydraulic, fossil and nuclear generating stations, Ontario Hydro operates 13 CANDU (CANadian Deuterium Uranium) nuclear units at two major multi-unit station site complexes (Fig. 1). Seven more nuclear units are under construction, including four at a third multi-unit site at Darlington. Ontario Hydro uses a systems approach in the development of its 'cradle-to-grave' management practices and facilities for radioactive materials arisings. The approach meets broad policy objectives set by Ontario Hydro with regard to management of radioactive materials to ensure an acceptable standard of health and safety for its employees and the public, and minimize the need for present and future commitments of human and materiel resources. With the prime objective of supply of electricity at the lowest feasible cost consistent with high safety and quality of service standards, a sound decision making approach in radioactive materials management is considered pivotal in the life cycle management of the materials, arising from its 14300 MWe (committed net electrical) nuclear generation program.

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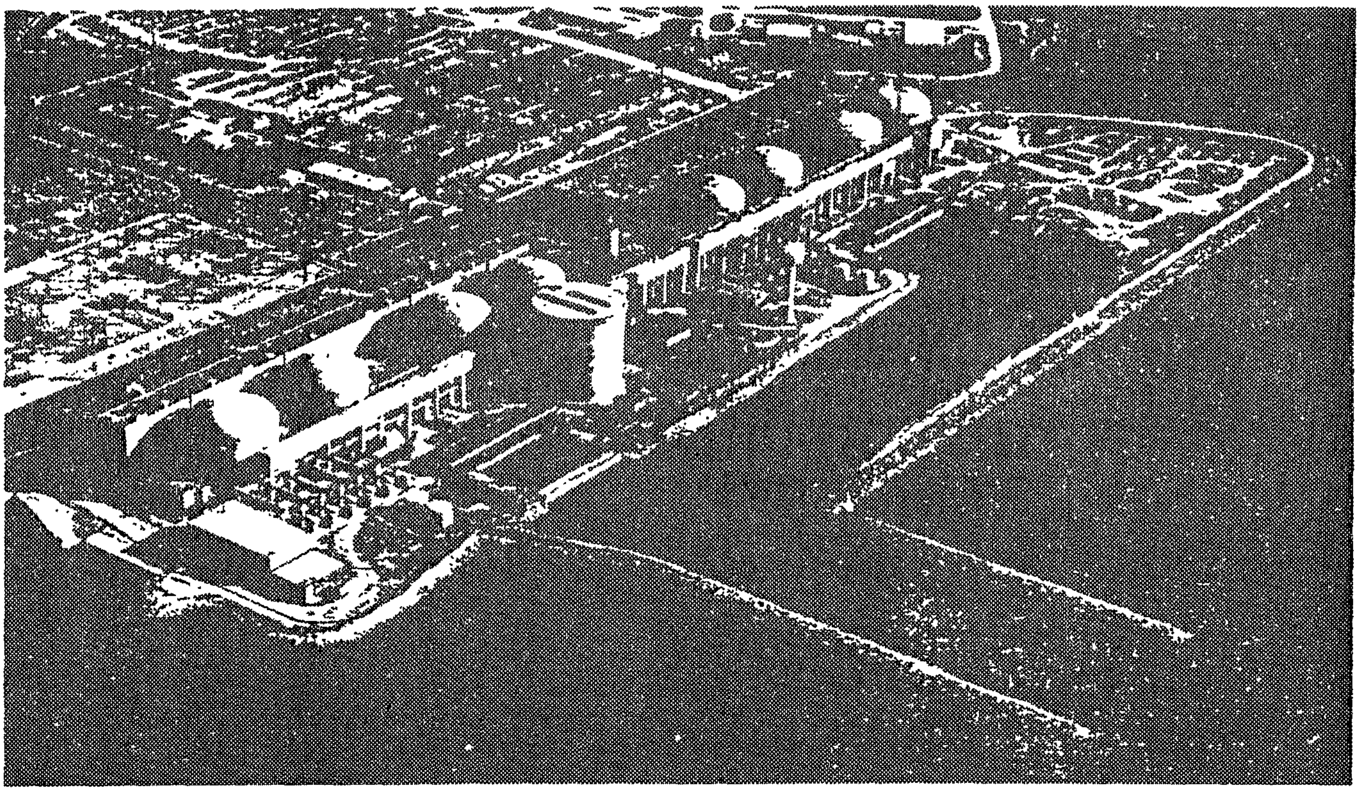

Fig. 1. Pickering Generating Stations A and B (Eight CANDU Units)

NATURE OF RADIOACTIVE MATERIALS

The radioactive materials generated from Ontario Hydro's CANDU nuclear generation program are diverse, both in physical character and radioactive hazard:

(a) *High Level Materials:* The largest quantity (99.9 percent) of radioactivity produced in the nuclear stations are contained within the irradiated fuel, generated as a result of the fission process. A typical four-unit CANDU station such as the one at Pickering Generating Station (GS) A discharges approximately 350 tonnes of irradiated fuel each year.

(b) Routine Reactor Wastes: These include process system wastes and contaminated housekeeping wastes from areas of the station where radioactivity is present. Although only a portion of the waste is actually contaminated with radioactivity, it is easier to treat all such wastes as radioactive. These wastes belong to two basic categories: low and intermediate.

Low level wastes comprise 95 percent by volume of all miscellaneous reactor wastes and include housekeeping materials such as paper and plastic sheeting, temporary floor coverings, used protective clothing (rubber gloves and plastic suits), mopheads, rags and other cleaning materials.

The more hazardous intermediate level wastes originate from the reactor process systems, and consist of impurities and particles in fluids passing through the reactor which have become radioactive. Organic ion exchange resins and filters are used to clean fluids in the reactor process systems, and require routine replacement. Metal parts, such as piping, valves and other hardware, and solidified liquid wastes, may fall into either low or intermediate level categories.

The volume of these wastes is relatively small compared to other typical industrial waste volumes. For example, a station the size of Pickering GS A (2060 MWe) produces about 1,000 cubic metres annually.

(c) Non-routine Wastes: These include reactivity control and flux measuring devices, such as liquid zone control rods, shutoff rods and flux detectors. These are extremely radioactive.

(d) Byproduct Wastes: These are wastes produced from the byproduct recovery operations. Ontario Hydro is the world's largest producer of Cobalt-60, a radioisotope widely used in cancer therapy and sterilization of medical supplies and agricultural produce. Although Cobalt-60 is the only byproduct currently marketed, systems are being developed for recovery of other marketable byproducts such as tritium and carbon-14. Wastes derived from the byproduct production will consist of highly tritiated solid wastes such as molecular sieves, charcoal filters and aqueous wastes such as electrolyzer fluid, pump oils as well as Co-60 and carbon-14 contaminated wastes.

(e) Irradiated Fuel Channel Components and Decontamination Wastes: Two units at Pickering GS A are currently under rehabilitation, which involves replacement of all fuel channels in the two cores. Each unit will discharge 390 sets of irradiated fuel channel components which include pressure tubes and fuel channel end-fittings. These components are highly radioactive at unit shutdown due to contributions from several short-lived radionuclides (Zr-95, Nb-95, Fe-55) and require interim management at the station sites. The retubing of the reactors is done after extensive decontamination, which generates significant quantities of intermediate level decontamination wastes, such as CANDECON (CANadian DECONtamination) resin (Co-60) wastes.

(f) Decommissioning Wastes: Although none of Ontario Hydro reactors will be decommissioned for many decades, the future dismantling of the nuclear structures is expected to produce significant quantities of irradiated components, decontamination wastes and low level contaminated structural materials.

SYSTEMS APPROACH IN RADIOACTIVE MATERIALS MANAGEMENT

The management systems for all the above materials have been evolving for the last fifteen years at Ontario Hydro. While most activities related to interim management such as waste characterization, collection and classification, processing and storage and transportation have already been developed, plans for long-term management of the irradiated fuel and the various wastes are being researched such that the Corporation's ultimate responsibilities for these materials can be discharged.

At every phase of development, detailed studies have been carried out, alternatives have been researched, lessons from previous experience have been reviewed and assimilated and decisions taken for new system components (e.g., facilities, processes and procedures). In this context, Systems Approach implies a thorough and systematic simulation of the identified alternatives at each decision point, allowing a comprehensive evaluation of all available optional courses of action and selection of the most appropriate option, considering cost, safety and various associated impacts. The policy objective of minimizing resource requirements and assuring acceptable health and safety impacts from radioactive materials management is thereby adequately assured in practice. The Systems Approach has not only left us a trail of sound decisions in radioactive materials management but also has provided a database consisting of a body of interrelated decision-support studies involving: nuclear generation experience and operating practices; collection, treatment, segregation, packaging and transportation; processing, volume reduction and interim storage; facility construction and operation; and disposal system requirements. The database is proving itself to be of immense value in the advancement of decision modelling to meet future requirements.

This approach has provided major directions to radioactive materials management over the last 15 years, from 1971 when Ontario Hydro, during the commissioning of Pickering GS A, established a centralized waste management facility at the Bruce Nuclear Power Development (BNPD) site (Fig. 2), and now includes a comprehensive program of facility/systems analysis and design; safety assessments and licensing; environmental and social assessments; siting and transportation routing assessments; and public/community studies, often carried out in cooperation with Atomic Energy of Canada Ltd, the universities and the private sector.

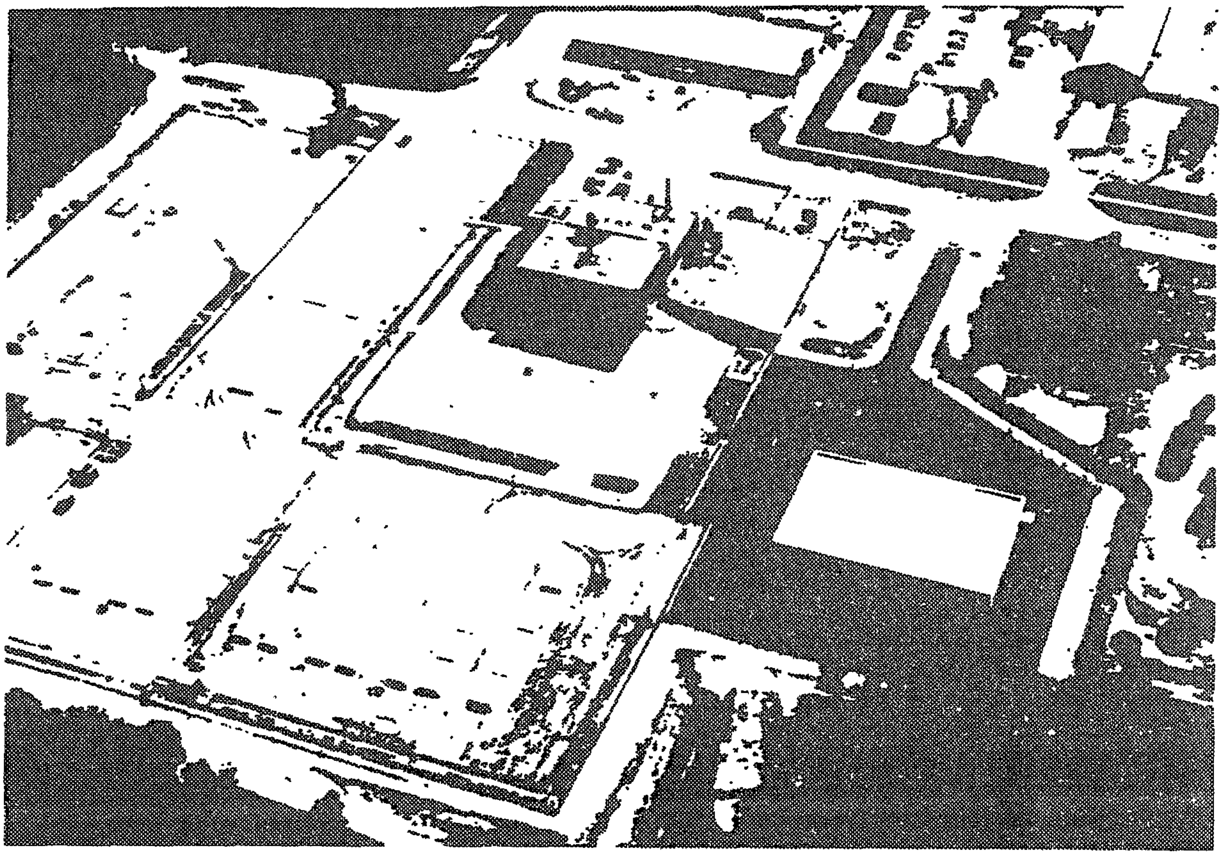

Fig. 2. Radioactive Waste Operations Site at the Bruce Nuclear Power Development

In the current socio-political climate regarding waste management, the balancing of technical and safety considerations with social and political needs is best achieved via this integrated approach.

SYSTEMS APPROACH METHODOLOGY

Although systems approach, from a nuclear technology transfer standpoint, has a number of areas that need to be tailored to a host country's specific requirements such as in the regulatory, infrastructure, industrial and socio-political contexts, the approach is largely generic. The methodology involves in a broad context an iterative process (Fig. 3) consisting of:

(a) Systems Analysis: evaluation and selection of alternatives, taking into consideration all activities and processes, interrelated from the point of view of feasibility, safety and environmental protection, etc.;

(b) Siting: evaluation and selection of sites and transportation routes from geological, material flow, technical and social/environmental standpoints, and;

(c) Facilities Engineering: evaluation and development of engineering and project related information for acquisition of systems.

Systems Analysis

Systems Analysis is a logical evaluation of management alternatives based on information on the waste arisings and forecasts, character of the wastes, processing alternatives, safety and licensing implications, and economic, siting and other constraints on the life cycle management of the wastes. It is an ongoing process and supports decision makers in the inception of programs and thereafter in identifying opportunities for improvement and reduction of costs.

A typical iterative stepwise process for the systems analysis of a radioactive materials management facility could be as follows:

(a) Forecasts and needs. Predictions of near-term and future waste arisings are somewhat difficult due to the uncertainties in generation growth, uncertainties in the station operations (such as outages, system upsets etc.) and the improvements generally incorporated at the stations in waste segregation, in-station handling and management of wastes. Nonetheless, predictions need to be revised on an ongoing basis to identify facility requirements, in-service dates and the throughput data for the design of the facilities. Some of the station operating conditions could have indirect effects on a wide range of waste parameters, (e.g., fuel defect rates impact on operations and maintenance waste quantity and activity levels). Throughput of different waste types could influence queueing requirements for transportation vehicles and the use of personnel for waste handling and processing. This part of the systems analysis is often termed "forecasts and needs assessments" and is often approached with detailed computational techniques. The forecasts and needs assessments are then used to generate overall material flow assessments which form the bases for study of alternative strategies, facility designs, transportation options, etc., and their interrelationships.

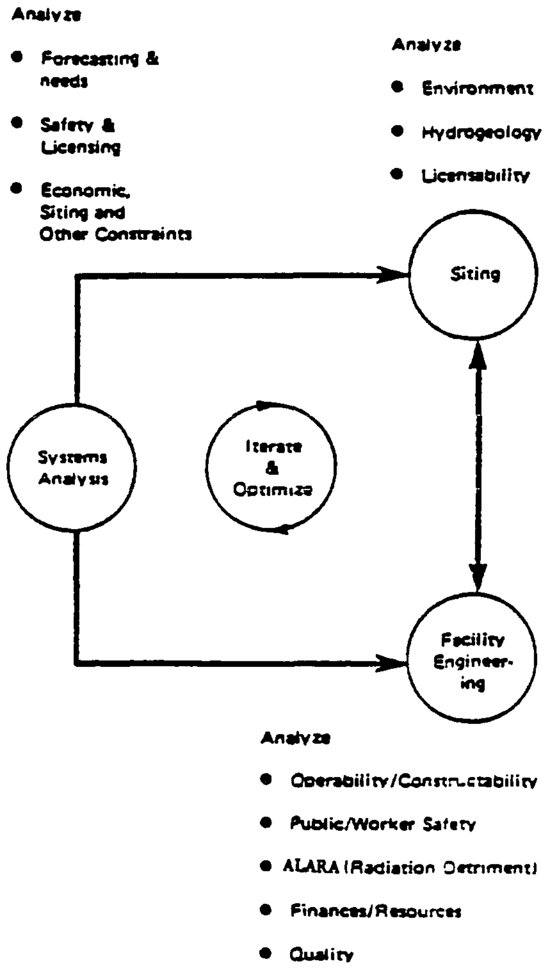

Fig. 3. Systems Approach

An example of a sophisticated tool for such systems analysis developed in Ontario Hydro is in the area of simulation and costing of irradiated fuel management strategies. By 1992, Ontario Hydro's nuclear generating program will consist of 20 units producing over 14,000 MW(e). Over 1,700,000 irradiated fuel bundles will have been generated by the year 2000. A computerized code called SCUFF[2] (System Costing of Used Fuel Facilities) has been developed to provide a tool to model the numerous irradiated fuel management strategies and parameters that are possible. SCUFF allows evaluation of possible variations to the nuclear generation program and to the storage, transport, and disposal phases. Major output from SCUFF include predictions of: irradiated fuel production; storage inventories and fuel flows; storage, transport, and disposal schedules and requirements; and detailed cost breakdowns for each phase.

In support of such computational simulations, detailed research and development, statistical and operational data are often assembled in database systems. One such system, referred to as Irradiated Fuel Management Data Centre (IFMDC) is in operation in the Nuclear Materials Management Department at Ontario Hydro.

(b) Safety/Environmental Assessment and Licensing. Identification of radioactive materials throughputs and their characterization provides the quantitative 'source term' information for systems analysis of safety and licensing aspects. These assessments include identification of all potential pathways of radioactive material to workers and the public and review of assessed dose and risks vis-a-vis regulatory limits and Ontario Hydro's radiation protection targets. The radiological pathways from the facility to humans is often complicated because of the numerous direct airborne and groundwater borne exposure pathways. The assessments not only include risks associated with life cycle management of the radioactive materials at the facility itself, but also include all associated 'systems' activities such as collection, segregation, packaging and transportation of wastes to and from the facility. Several state-of-the-art computer pathway models, acquired, as well as developed in-house, are now available for simulating the radiological pathways.

Environmental evaluations of the systems are concerned with potential impacts from construction and operation of the facility as well as associated activities such as transportation. The environmental effects are reviewed against criteria based on regulatory guidelines for environmental protection. These criteria cover a large number of areas such as air quality; water quality; protection of aquatic life; protection of vegetation and wildlife; protection of natural and historic features; land use/capability, effects on non-renewable resources; etc. A number of social impact factors are also assessed, such as in the areas of demography and economy, effects on the social environment, public services, local and regional planning and in general, the overall effects of the facilities on the community and the public.

(c) Economic. Detailed economic analysis of alternatives are carried out to screen the financially preferred alternatives. The alternatives need to be described with:

(1) Material flows, i.e., quantities and types to be dealt with as a function of time.

(2) Development of schedules for construction/operation/retirement of facilities to match the material flows.

(3) Development of logistics of transportation options.

(4) Identification of all remaining resources necessary for bringing an alternative into service.

It is also necessary to optimize each alternative for cost reduction by minimizing idle capacities, timing of expenditures and optimizing the various construction/operation/closure activities for economic efficiency.

The financial characteristics of the alternatives are assessed in Ontario Hydro against four major financial criteria:

- Internal Economic Efficiency
- Affordability
- Corporate Impact
- Uncertainty

Internal Economic Efficiency is measured by the Net Present Value (NPV) of each alternative expenditure. The lower the NPV, the greater the contribution to the internal economic efficiency.

Affordability is measured by the financial resources required in the short-term over those in the long-term and is important, especially at times of economic constraint. The lower the short-term resources required, the more affordable the alternative.

Corporate Impact is measured by the revenue requirements (i.e., impact on bulk power rates), initial budget requirements for development of new initiatives, changes required to the Corporation's financial statements and impacts on financial objectives (debt/equity ratio), policies and strategies.

Uncertainty evaluation takes into consideration the variability in the economic assessment due to a number of potential factors such as lack of reliable estimates, potential changes in technology, regulations, etc.

In summary, Systems Analysis provides on the one hand, a comprehensive database in which alternatives are assessed from technical, economic and safety/environmental/social perspectives and on the other, a powerful decision modelling tool for major management decisions.

Siting

The next "program centre" in Systems Analysis is Siting. The siting of radioactive materials management facilities is an increasingly complex process because of the increasing desire for public/community involvement in decision making processes.

The Bruce Nuclear Power Development (BNPD) Site was a natural choice for the siting in 1971 of the centralized waste management site because of its proximity to the (then future) Bruce generating stations thus taking advantage of resources associated with the stations such as construction infrastructure, a pool of trained manpower skilled in the handling of radioactive wastes, health physics programs, and environmental radiation monitoring programs. These features contribute to minimum costs for both waste operations and overall system transportation. The site has various other favourable characteristics such as land availability for future development, distance from large centres of population, distance from sources of water used by, or accessible to the general public, low seismic activity, functional and controlled year round access, and a good hydrogeologic setting.

For new siting programs such as those that may be required for hosting future facilities, capabilities are available for the development of siting methodology, environmental assessment, hydrogeological characterization and site specific safety and licensing studies.

(a) Siting Methodology. Site selection will likely be the most difficult and controversial component of planning for a facility because of the public's negative attitude against hosting it in their community (Not-in-my-backyard syndrome). Due to the contentious nature of siting a facility, it will be necessary, at some appropriate point, to open up the siting process to public and regulatory review. Therefore, the process followed must be well-thought-out, logical, flexible enough to incorporate new data/concerns, and readily defendable from public inquiry at any level of detail. Technically, the process used to select a site for a facility must consider the total waste management system, including transportation, processing/ conditioning, storage, caretaking, safeguards, (security) monitoring and the operating life of the facility and facility retirement. It must also emphasize overall objectives of minimizing program costs, minimizing potential public health hazards, while maximizing environmental protection.

The site selection approach which has evolved from generating station and transmission line siting generally follows a procedure of study area screening and evaluation utilizing a series of techno-economic, environmental and social criteria. More detailed criteria and data are utilized at each successive stage of screening.

The process (Fig. 4) involves:

(1) Identifying a Candidate Region.

(2) Identifying Candidate Areas within that Region.

(3) Identifying Candidate Sites within Candidate Areas.

(4) Evaluating Candidate Sites to identify Preferred Site(s).

(5) Evaluation of Preferred Site(s) for hosting facility Concept Alternative(s).

In the initial stages of the process, criteria will involve broader issues such as transportation, hydrogeology, exclusion from built-up areas and dedicated lands (Indian reserves, national parks etc). As the screening progresses, more detailed criteria in techno-economic, social/community, natural environment, land use and transportation areas are brought to bear.

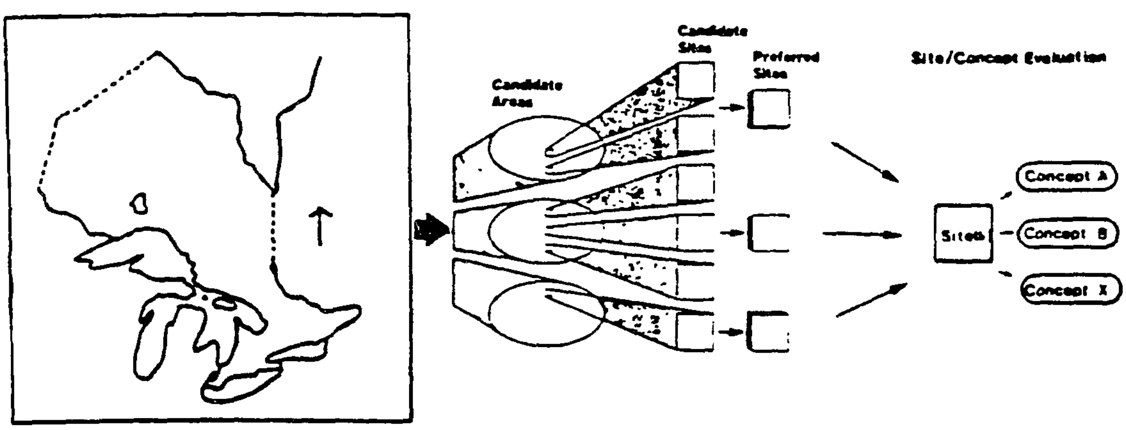

Fig.4. Site Selection Approach

(b) Environmental Assessment. The Environment Assessment Act of Ontario (referred to as EA Act) requires proponents of proposed undertakings to submit an environmental assessment to the Minister of the Environment. In the siting of new facilities, Ontario Hydro is subject to this Act.

The objectives of the EA process[3] are:

(1) To identify and evaluate all potentially significant environmental effects of proposed undertakings at a stage when alternative solutions, including remedial measures and the alternative of not proceeding, are available to the decision-makers; and

(2) To ensure the proponent of an undertaking, and Governments and agencies required to approve the undertaking, give due consideration to the means of avoiding or mitigating any adverse environmental effects prior to granting approval to proceed with the undertaking.

In addition the Act imposes other requirements. One of the most important requirements is public involvement. Formal provisions of the statute require:

- Provision of adequate public notice for major events in the process prior to the events taking place;

- Access to information by maintenance of a Public Record file containing all pertinent information in relation to the undertaking and the government review;

- Provision for formal hearings which can be requested by any member of the public.

Although not required by statute, the Ministry of the Environment (MOE) also encourages proponents prior to EA submission, to participate in a "presubmission consultation review" where affected government ministries/agencies as well as the public provide assistance in scoping the EA document, in identifying areas of interest or concern, identifying mitigation alternatives and if possible, directly contributing to project decision-making (e.g. siting, selection of alternatives etc). Experience confirms that most major problems or differences that can be resolved, are best dealt with during the EA presubmission consultation. This may avoid hearings or lessen the atmosphere of confrontation common to hearings.

(c) Geological/Hydrogeological. The site selection process would go through a series of geological/hydrogeological investigation stages which progressively eliminates less favourable sites and focus on the sites that are considered most suitable. As the investigations progress through each stage the activities evolve from collecting preliminary and descriptive data to activities which collect more specific and quantitative data. During the final site confirmation stage the hydrogeologic site investigation activities would be very detailed. The results would be used to develop engineering specifications and detailed cost estimates for the final design of the facility.

Site confirmation would not end with field investigations but would continue throughout the construction and operation, shutdown, sealing and surveillance of the facility. For instance during the excavation work for the facility, observations of the soil or rock would be made to determine the accuracy of earlier site investigations, and design changes would be made accordingly. In some cases it may be advantageous to perform in situ investigations as the soil or rock is exposed during excavation. Following the closure of the facility, there would be a period of monitoring to assess facility performance and provide final confirmation that the appropriate site and facility design have been chosen.

In the geological screening of the sites, criteria are established in the areas of ground/surface water characteristics, geochemical properties, geomechanics and tectonics, etc. The geological criteria become particularly important for facilities that rely on geological barriers as the mainstay of safety.

Ontario has superior bedrock groups, Paleozoic and Pre-cambrian in age and surficial geology which includes till plains and moraine regions consisting of cohesive soil types. Over the years, a detailed database and a collective expertise has been acquired with the support of University of Waterloo Department of Earth Sciences.

Ontario Hydro has researched[4] a range of site investigation techniques over the past several years not only in support of the waste operations site at BNPD, but in general, in support of various generation and transmission activities of the utility. These techniques include (a) soil sampling techniques to define stratigraphy, (b) hydraulic head and conductivity measurements with piezometers, (c) analysis of groundwater samples for major ions, trace elements and for naturally occurring isotopes to determine groundwater age and origin, (d) regional hydraulic tests by well-pumping techniques, (e) tracer tests to estimate in-situ values of dispersion and Kd of radionuclides in the soil, (f) laboratory batch K_d measurements to complement tracer studies, and (g) laboratory studies to quantify hydrogeochemical processes affecting radionuclide midgration in the geomedia. This experience is continuing to lend support for on-going site investigations at the Bruce waste management site as well as for acquisition of a generic database on Ontario geology.

(d) Site-Specific Safety Modelling

The performance assessment modelling using site specific geologic and hydrologic data is an important element in the evaluation of sites/facilities especially those that involve geologic storage and disposal. Computer models which use finite element techniques are often used in the performance assessment of concepts. These mathematical models allow evaluation of the relative importance of various site specific hydrogeological and engineering parameters which affect the release and migration of radionuclides from the waste.

The groundwater pathway often receives much attention during performance assessment modelling studies. The mobilization of radionuclides by water and the subsequent release to the groundwater environment has a significant probability of occurring relative to other release scenarios. To predict radionuclide migration through the subsurface, site specific mathematical models must be employed which simulate the advective, dispersive and geochemical processes acting on the radionuclides. In order to use these models, groundwater velocity distributions and data describing the dispersive and sorptive characteristics of the geologic material obtained from hydrogeological characterization are input into the models.

Facilities Engineering

Facilities Engineering makes use of the information generated in Systems Analysis and Siting studies and generates design alternatives for the facilities and processes. Manufacturing support and consulting assistance from leading firms in the private sector, such as London Nuclear, the proponents of CANDECON decontamination technology, is called upon as required. Operability/constructability, public and worker safety, licensability, commitment to ALARA, and economics require an iterative approach in the design process. Corporate requirements on financial evaluations, quality, budgeting and resourcing have to be taken into account. Generally, design of systems as 'modular' units, repeatable as required, reduces one-time cash requirements and improves affordability.

When the study of alternatives is completed and decisions for acquisition of systems are taken, preliminary engineering and detailed design of the recommended alternative is taken up. This process includes, to the degree warranted, the preparation of:

(a) Project performance specifications containing system requirements and plant performance specifications;

(b) Safety design guides outlining the safety design procedure for the facility;

(c) Project environmental requirements, identifying environmental design criteria to be followed, based on environmental regulations and on any other conditions imposed arising from the environmental assessment process by government ministries based on their analysis of environmental assessment and reported public concerns;

(d) Project reliability and maintainability requirements;

(e) Design requirements and descriptions, quality engineering plans; and

(f) Detailed design.

TWO CASE STUDIES OF SYSTEMS APPROACH

Two cases where Ontario Hydro carried out in-depth studies using a Systems Approach can now be described. These are the (1) irradiated fuel storage siting study and the (2) irradiated components management systems study.

Irradiated Fuel Storage Siting

The irradiated fuel discharged from Ontario Hydro's nuclear generating stations is presently stored in waterfilled pools at the stations. This method of storage is used worldwide and has proven to be safe, economic, and reliable. Eventually, the irradiated fuel will either be disposed of directly as a waste or be reprocessed to recover the fissile material remaining in the fuel. A decision on whether or not to reprocess the irradiated fuel has not yet been made in Canada. The disposal concept, presently being studied in Canada is to immobilize either the irradiated fuel or the high level radioactive waste from reprocessing in durable containers and emplace them in a mined cavity 1000 metres below the surface in plutonic rock of Precambrian origin. Until disposal, the irradiated fuel will continue to accumulate and additional storage facilities will be required. Storage of irradiated fuel in waterpools offers an excellent management method for many decades.

A detailed systems study[5] evaluated the options available to Ontario Hydro, in the interim period, for the siting of the irradiated fuel storage facilities.

The major siting options considered were:

(a) Extending the storage of irradiated fuel at the station sites.

(b) Storage of irradiated fuel at a central site, the facility located at:

   (1) An independent site,

   (2) An existing site, such as the Bruce Nuclear Power Development (BNPD) site.

   (3) The disposal site.

The study reviewed in depth the following:

(a) Reference designs for the required facilities.

(b) Material flows between the places of origin (generating stations), storage pools and the final destination (repository): development of a strategy for transportation of irradiated fuel between facilities.

(c) Feasibility of siting of the facilities and engineering of the scenarios: assessment of the impact of the scenarios on Ontario Hydro's existing facilities in terms of constructability and interfacing with operating facilities.

(d) Effect of the scenarios on public safety, occupational safety, environmental quality and communities.

(e) Difference in costs between scenarios.

The study recommended that Ontario Hydro plan to continue storing the irradiated fuel on-site (i.e., at the stations) until the decision on whether or not to reprocess the irradiated fuel has been made; if it is decided not to reprocess the irradiated fuel then it should continue to be stored at the station site until the disposal facility becomes available.

The systems studies are continuing to assess a number of avenues for cost reduction and long-term advantages; these include various modular dry storage techniques and integrated systems for once-through handling of fuel from storage to disposal. No decisions, however have been arrived at for changing course either from on-site storage to centralized systems or from waterpool concepts to dry concepts for storage of fuel. However, it is recognized that the cost of providing an irradiated fuel/fuel waste immobilization, packaging and disposal facility is high and the economic studies show that the total irradiated fuel management costs including extended storage and transportation components, continue to significantly decrease by deferring disposal to at least the year 2050. It is also recognized that centralized extended storage is as viable as continued on-site storage at the various nuclear station sites and has advantages from optimization of a central facility design and from elimination of caretaking and construction impacts imposed on stations.

Irradiated Components Management Systems

What do you do with 7 metre long irradiated fuel channel assemblies having dose rates of up to 150 Sv/h (15,000 R/h) after they are removed from the reactor? The Unit 1 and 2 fuel channel replacement program for Pickering GS A will generate 780 of these assemblies along with 780, 2 metre long irradiated end fittings during the period 1985 to 1987. Prior to undertaking this replacement program, Ontario Hydro facilities for transporting and storing these materials were assessed to be inadequate because of the high activity, physical dimensions and total overall volume of the assemblies. Existing Pickering irradiated fuel bays could possibly have been used to store the assemblies, but besides creating conflicts with the normal day to day operation, storage in the bays would only have been feasible until the early 1990's at which time all the remaining space in the pools would have been needed for storage of irradiated fuel. Delays in the retubing program would further shorten the available storage time in the bays.

New facilities, therefore, were required to transport, handle, process and store the assemblies. Several different irradiated component management scenarios were considered taking into account, among others, the following factors:

(a) Occupational and public safety

(b) Compatibility with existing Ontario Hydro practices

(c) Compatibility with Ontario Hydro Waste Management Policies

(d) Man-rem expenditure targets

(e) Compatibility with the overall replacement program

(f) Cost

(g) Flexibility to cater to future changes in regulations affecting storage

(h) Developmental problems

(i) Other benefits not directly related to cost

The management scenarios included seven on-site and off-site storage options with and without volume reduction. Storage methods included new on-site waterpools, a new horizontal bunker type facility and a vertical storage type facility similar to that used for storing ion exchange resins at BNPD, and use of existing pools such as the irradiated fuel bays and the fuelling machine maintenance bays.

Large shielded concrete bunkers called 'Dry Storage Modules' (DSM's) were finally selected as the optimum storage method, with a temporary on-site storage strategy. The system[6] includes a fuel channel flask for handling the assemblies from reactor face to a transfer station and dry storage modules, each a 153 Mg concrete and steel transferrable/transportable shielded container (Fig. 5) designed for once-through handling of the irradiated components through various phases (storage to disposal) in the management of these materials.

A REVIEW OF THE DECISION TRAIL IN ONTARIO HYDRO'S RADIOACTIVE WASTE MANAGEMENT

Some of the major decisions and resulting facilities in the long trail of Ontario Hydro's radioactive waste management can now be summarized at least in two major areas not discussed earlier:

Solid Wastes

For solid wastes other than irradiated fuel, mostly low level maintenance wastes, studies identified a major need for effective volume reduction; a centralized processing and storage facility was therefore assessed to be the most economic. A prototype incinerator for processing more than half the bulk waste was introduced in 1977 along with a compactor/baler system for the remaining compactable wastes. The wastes, currently amounting to about 5000 m^3/a, are transported by road to the central waste management facilities using Ontario Hydro LSA, Type A and Type B packages, developed specially to match waste characteristics.

These facilities located at the Bruce site have been effective in keeping our total waste quantities to manageable levels; these wastes are stored in in-ground and above-ground engineered storage facilities. Ongoing studies are also in progress for further reducing these volumes by supercompaction and by declassification, i.e. better sorting systems to segregate wastes that could be disposed as non-radioactive.

Systems studies are currently pointing to a need for reorientation of the radioactive waste management program towards disposal in the years to come. This is to improve the economy of radioactive waste management in the face of the maturing nuclear generation program and also to meet corporate responsibilities with respect to long-term management of radioactive wastes.

Facilities currently in operation are perhaps the most 'visible' area of Ontario Hydro radioactive waste management, and are considered the forerunners for 'turn-key' projects involving technology transfer. These systems and facilities have evolved from an ongoing systems approach to decision making.

The major facilities include: a volume reduction facility, consisting of a large pyrolysis type starved air incinerator (Fig. 6), a compactor and a baler; in-ground facilities such as concrete trenches and tile holes for low and intermediate level wastes (Fig. 7); storage buildings (Fig. 8) used for above-ground storage of low level wastes; "Quadricells"

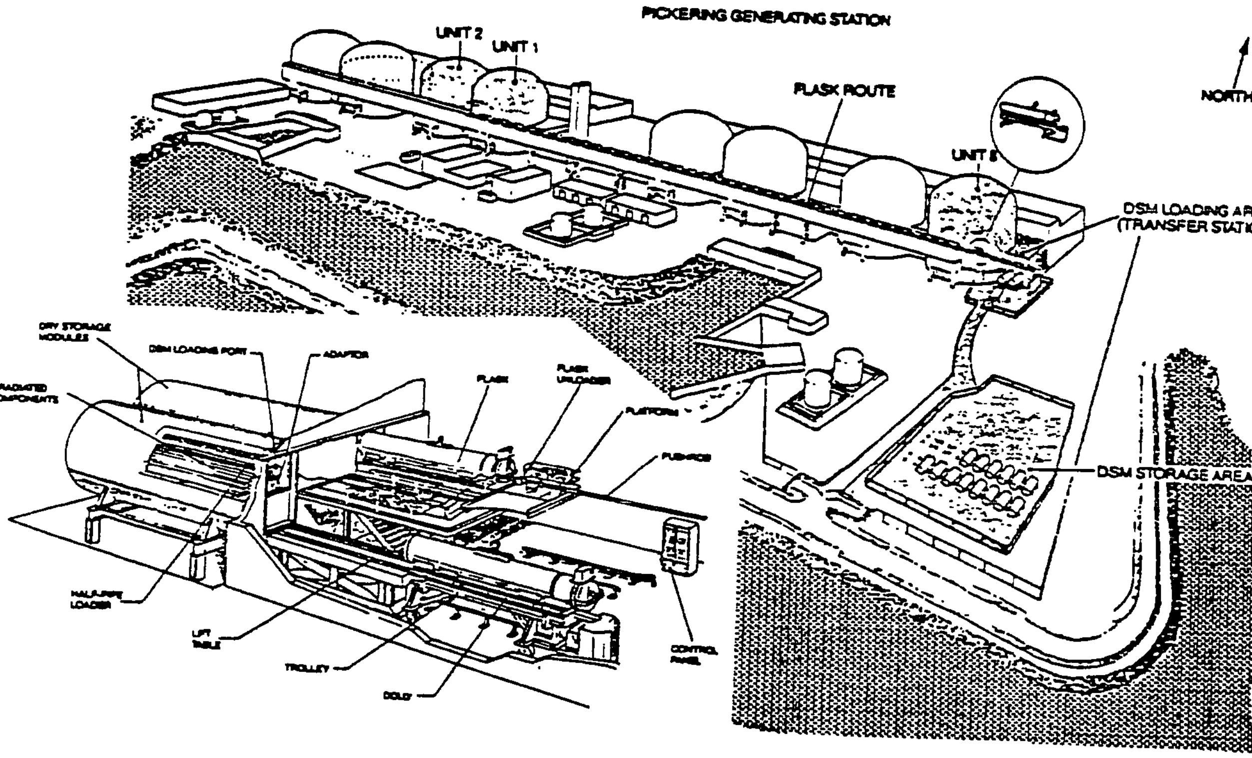

Fig. 5. Dry Storage Module

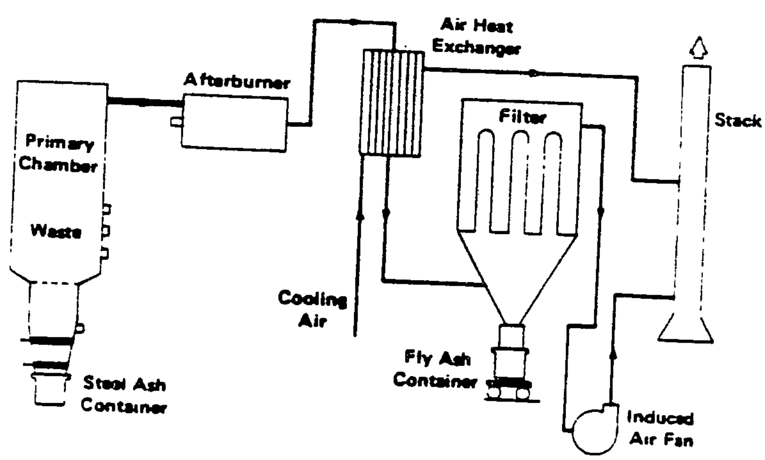

Fig. 6. Incinerator Schematic

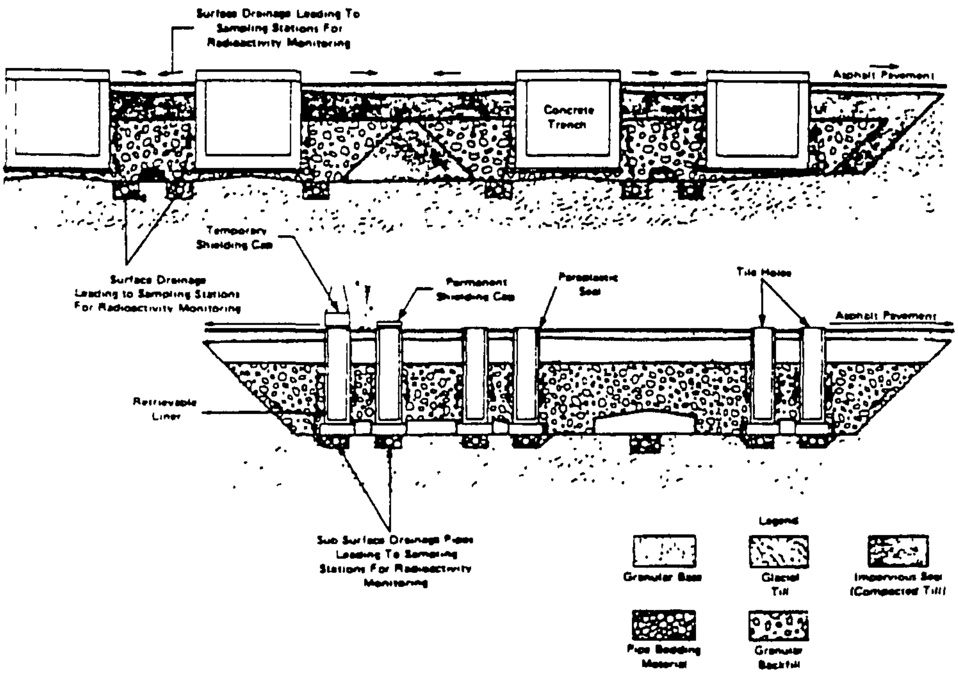

Fig. 7. Trenches and Tileholes

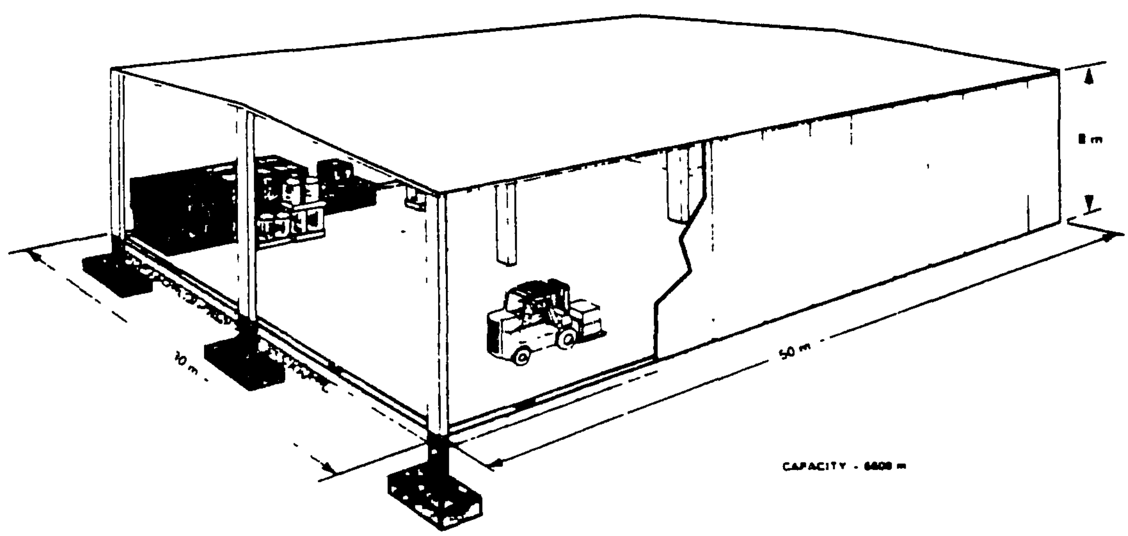

Fig. 8. Low Level Storage Building

for above-ground storage of highly radioactive waste components such as resins (Fig. 9); dry storage modules (DSM's) which are transferable/ transportable concrete containers for highly radioactive reactor core retubing wastes which were discussed earlier in the case study; and custom-made on-site and off-site transport packages, meeting national and IAEA transportation regulations where applicable.

The design philosophy for these facilities has evolved over a period of fifteen years(7), owing largely to the reviews based on a systems approach at each stage of acquisition of new facilities and operational experience feedback into this process over the years.

Several new storage and processing facilities have also been brought into service in the recent past. Among these are (i) the in-ground storage containers, which provide storage capacity for intermediate level wastes and are developed to minimize the excavation and concreting required to construct the facility, utilizing vertical borehole augering techniques (Fig. 10); and (ii) Radblocks, another flexible intermediate level waste management concept (Fig. 11) under prototype development; Radblock is a portable, pre-cast reinforced concrete structure provided with four or five internal cavities in which waste components are placed. The concrete provides radiation shielding and provides engineered containment to the waste, while the package is used for storage, transport and disposal.

Tritium

Another area where a centralized system of management has been preferred is tritium removal. It happens that Canada is also a major producer of tritium(8). This isotope of hydrogen is produced in the Ontario Hydro nuclear reactors by the bombardment of the deuterium in heavy water by neutrons. Tritium is radioactive and its concentration builds up with time in the moderator and coolant systems and contributes to the radiation dose of station workers.

Technologies have been developed to remove tritium from these reactor systems, but what is a nuisance to station operators is also a fusion fuel, and offers a possibility for commercial development. CANDU's will never produce enough tritium to fuel commercial fusion reactors; these reactors will manufacture their own. However tritium will be required in substantial amounts for advanced fusion experimental reactors and for the startup of commercial fusion reactors when they are built.

The complex systems (catalytic exchange/cryogenic distillation) required for tritium removal (Fig. 12) justified the decision for a centralized tritium removal facility, now under construction at Darlington GS. Tritium from this facility will be marketed as a fuel for fusion research and other peaceful commercial applications. The tritium byproduct will be produced in a solid form bound as a metal tritide in unit sizes up to 500,000 Ci.

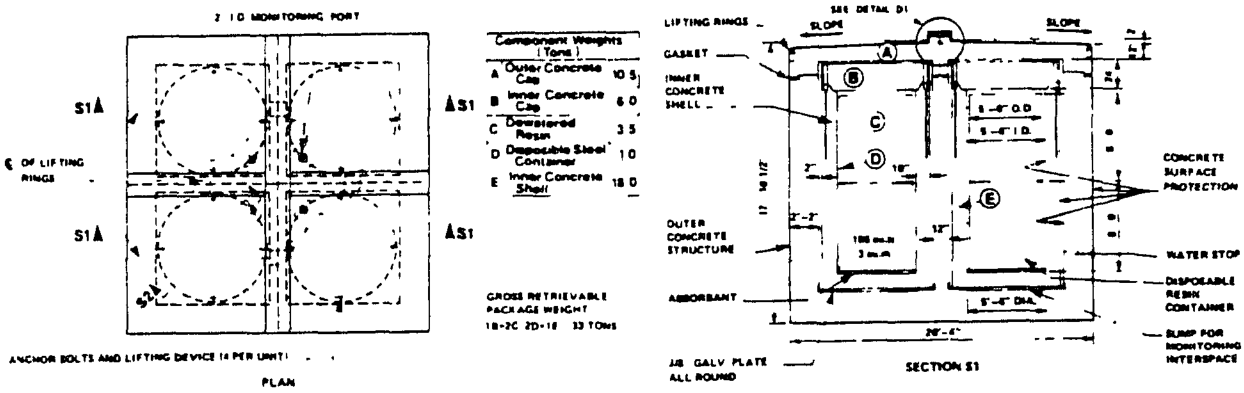

Fig. 9. Quadricell

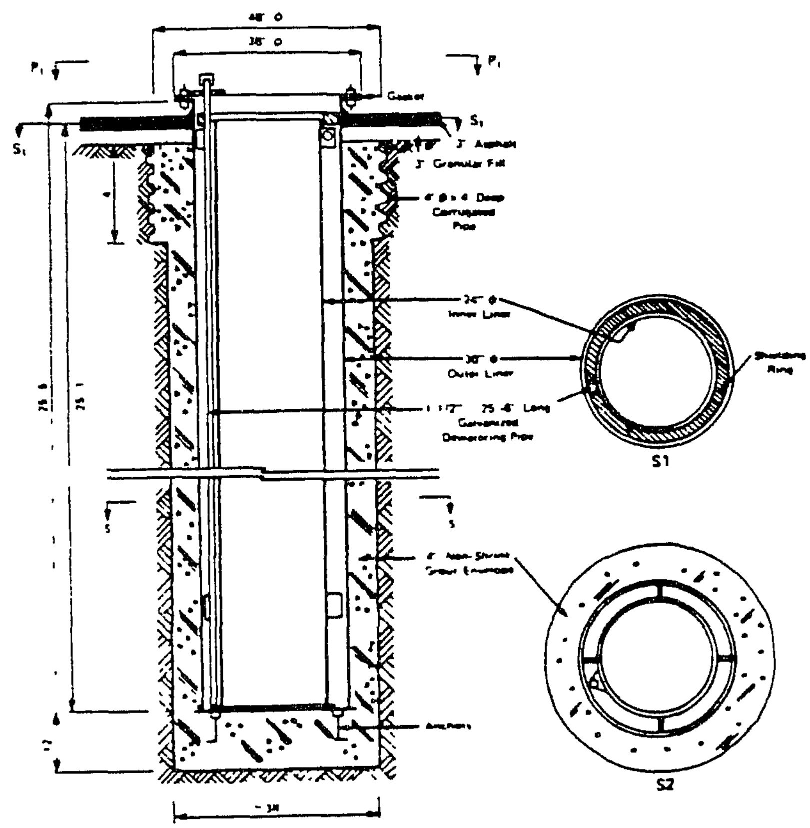

Fig. 10. In-ground Storage Container

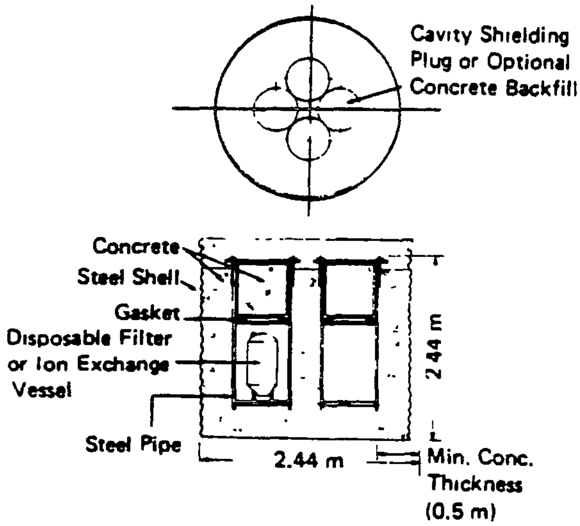

Fig. 11. Radblock

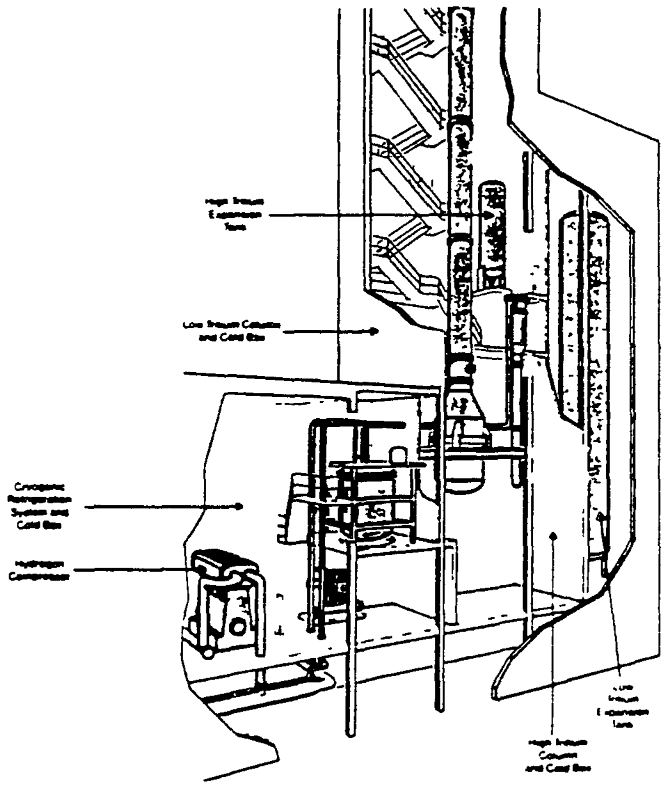

Fig. 12. Tritium Removal Facility

CONCLUSIONS

Ontario Hydro has enjoyed considerable success during the development of its radioactive materials management systems based on the use of "Systems Approach" in decision making.

Highly resourced programs are now in place to take the utility through the challenges of developing new facilities to meet the needs of a 14300 MWe nuclear generation program.

The major strength of the Systems Approach is its ability to provide proper decision support, often in the face of a host of diverse alternatives.

In a large system such as a nuclear-oriented utility's radioactive materials management system, a System Approach requires (i) an ongoing review of accumulated experience, (ii) a research and development database to lend support to informed decision-making and most importantly (iii) an iterative process of Systems Analysis, Siting and Facilities Engineering such that optimum decisions emerge, compatible with the technical and social environments.

ACKNOWLEDGEMENTS

Although overall responsibility for development of radioactive materials management systems lies with the Nuclear Materials Management Department, many departments have contributed to the development of the Systems Approach. These include Ontario Hydro Research Departments, Environmental Studies and Assessments Department, Geotechnical and Hydraulic Engineering Department and the various operations, project and design departments. These contributions are gratefully acknowledged.

REFERENCES

(1) Ontario Hydro 1984 Annual Report, copies available from Ontario Hydro Head Office, 700 University Avenue, Toronto M5G 1X6.

(2) J.M. Cipolla, "SCUFF: A Computer Code for Simulating and Costing Used Fuel Management Strategy", Paper Presented at the International Topical Meeting on High Level Nuclear Waste Disposal, Washington, USA - September 24 - 26, 1985.

(3) Guidelines for the Implementation of the Environmental Assessment Act in Ontario Hydro - Generation. Ontario Hydro Environmental Studies and Assessments Report No. 84113, June 1984.

(4) R.J. Heystee and P.K.M. Rao "Canadian Experiences in Characterizing Two Low-Level and Intermediate Level Radioactive Waste Management Sites", Ontario Hydro Design and Development

Division Report No. 84132, April 1984. Presented at the IAEA Seminar on the Site Investigation Techniques and Assessment Methods for Underground Disposal of Radioactive Wastes, Sofia, Bulgaria, February 1984.

(5) "Management of Irradiated Fuel: Storage Siting Options", Ontario Hydro Design and Development Report No. 79418, December 1979.

(6) I.E. Wall and Z.S. Beallor "Management of Irradiated Components For The Pickering Units 1/2 Retube", Paper presented at the CNS/CNA Conference, Ottawa, June 1985.

(7) T.J. Carter and P.K.M. Rao "Fifteen Years of Radioactive Waste Management at Ontario Hydro", Paper Presented at Waste Management '85, Symposium on Waste Management at Tucson, Arizona, March 24-28, 1985. See Proceedings, Vol 2 pp 445-451.

(8) "Fusion Energy and Canada's Role", Canadian Fusion Fuels Technology Project (CFFTP) publication, copies available from Dr. T.S. Drolet, Canadian Fusion Fuels Technology Project, Ontario Hydro.

BASIC PRINCIPLES ON THE LIMITATION OF RADIATION DOSES SUPERVISION AND WARNING SYSTEMS

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Due to the recommendations of the International Commission on Radiological Protection (ICRP - Publ. No. 26/1977) and the International Atomic Energy Agency (Basic Safety Standards for Radiation Protection, Edition 1982) and the further development in this field, it is obligatory for all states, in the case of handling with sources of ionizing radiation, as e.g. X-ray equipment, radioactive substances, fissionable materials, nuclear reactor plants, installations for the storage of nuclear waste, reprocessing plants, waste conditioning plants and underground plants for final radioactive waste disposal, particle accelerators, etc., to have a rigorous legislation for licensing in the field of radiation protection and to supervise with adequate institutions, personnel, methods and instruments:

(a) the quality of all radiation sources;

(b) the extent and dose rates of radiation fields from diagnostic and therapeutic medical equipment concerning the radiation doses of patients and workers;

(c) the extent and dose rates of radiation fields from non-medical X-ray and gamma-ray equipment concerning the radiation doses of workers;

(d) the laboratories dealing with radioactive substances;

(e) nuclear plant projects prior to and during construction including controls of all safety aspects of the site, building constructions, all systems and documents for safe operation, the aspects of physical protection and the concepts for storage, conditioning and final disposal of radioactive waste and for dismantling, storage and final disposal of contaminated and activated radioactive components of these plants, and, last but not least, the quality of personnel;

(f) the concepts concerning early notification and protection measures in the event of accidents;

(g) the external dose rate including the natural background level, nuclide specific radioactivity in air, water, soil, food and agricultural products in the neighbourhood of nuclear plants and in a supervision-network distributed over the whole state;

(h) the function of all safety systems in the plant and warning systems in the plants and in their neighbourhoods;

(i) all aspects of safe operation including the quality of personnel;

(j) the dose equivalents and the committed effective dose equivalents by external radiation and external contamination by inhalation and by ingestion of radioactive substances concerning:

  (i) personnel for operation, maintenance, training, auxiliary and emergency action and office staff;

  (ii) individual members of the public and critical groups of the population; and

  (iii) the whole population of a state;

(k) the conditions for the safe transport of radioactive substances, physical protection measures in the case of fissionable material transport and in the case of transport of components which can be used for the construction and the application of critical fission assemblies;

(l) that all fissionable materials are always really under safeguards control by the IAEA.

In trying to get more benefits for one's state by the use of ionizing radiation, one has to consider that risks to people will rise.

The use of ionizing radiation seems tolerable if the benefits for a certain population group are higher than the risks (also for this group) and if other methods with lower risks for the same purpose do not exist. But this philosophy is only true to a certain extent because one cannot tolerate higher and higher risks to get higher and higher benefits. Therefore, one has to set limits not only for the nonstochastic but also for the stochastic effects in the population or in critical groups, e.g. for the group of occupationally exposed persons, concerning somatic and genetic effects.

In this connection, it is necessary to keep in mind that it is not only radiation risks which exist but also, after a decision to build up for example some nuclear plants, financial, economic, social, cultural risks and risks of various kinds of dependence can arise. Therefore, it should be considered:

(a) whether the benefits are really greater than all these risks; and

(b) whether these risks are not too high.

As far as radiation risks and radiation dose control are concerned, it is necessary to limit the sum*/ of:

(a) the dose equivalents or whole body doses or effective dose equivalents by external radiation;

(b) the doses correlated by the activity intakes of radioactive substances via the gastrointestinal tract (oral intake); and

(c) the doses correlated by the activity intakes of radioactive substances via the respiratory tract (inhalation intake).

The tolerable amounts for oral intake, inhalation and external radiation doses including radiation influences after nuclear accidents are correlated due to the following formula for members of the public:

$$\frac{\sum H_{wb\ dose\ equiv.}}{0.5\ \text{millisievert}} + \sum_i \frac{A_{i,\ oral}}{1/100\ ALI_{i,oral}} + \sum_n \frac{A_{n,\ inhal}}{1/100\ ALI_{n,inhal}} \leqq 1$$

$\sum H_{wb\ dose\ equiv.}$ — the sum of all during one year received whole body dose equivalents in parts of millisievert;

$\sum_i$ — sum of quotients concerning the different toxic radioisotopes related to ingestion

$A_{i,oral}$ — sum of activity intakes during one year by ingestion, concerning the i^{th} radioisotope
unit: Becquerel

$ALI_{i,\ oral}$ — annual activity limit on intake by ingestion concerning the i^{th} radioisotope for occupational workers
unit: Becquerel

$\sum_n$ — sum of quotients concerning the different toxic radioisotopes related to inhalation

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*/ After the Chernobyl accident, it seems no longer correct to set limits only in connection with point (b).

$A_{n,\ inhal}$ — sum of activity intakes during one year by inhalation concerning the n^{th} radioisotope
unit: Becquerel

$ALI_{n,\ inhal}$ — annual activity limit on intake by inhalation concerning the n^{th} radioisotope for occupational workers
unit: Becquerel

Concerning oral intake and inhalation of radionuclides, account must be taken of differences in organ size and metabolic characteristics of infants and children.

Influences on the skin and via the skin should be regarded independently from the above-mentioned formula.

For occupational workers, lower values for annual dose limits than those which are presently recommended should be considered (about one-fifth or three-fifths of the present values).

In 1978, Austria decided against the production of electricity by nuclear power reactors; three research reactors, however, are in operation.

Austria has at its disposal a modern monitoring system to measure, permanently, external radiation dose rates.

A telecommunication system brings these data on-line to central units where the information is registered.

Many institutes of the universities, hospitals and industries are working with radioactive substances, X-rays, gamma-ray equipment and accelerators in medical and non-medical fields.

Austria has a lot of experience in medical and technical applications of ionizing radiation and will be pleased to submit this knowledge to other states on request.

RADIATION PROTECTION IN SWEDEN: PRINCIPLES AND PRACTICE

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The radiation protection act

The present act concerning protection against radiation was passed in 1958 and replaced the act from 1941. It regulates possession of and work with radioactive substances and x-ray equipment or other technical devices designed to emit ionizing radiation, work at nuclear installations and trade with and transfer of radioactive substances. The provisions of the act can also wholly or in part apply to non-ionizing radiation.

The present Swedish radiation protection act is mainly intended for protection of workers. In 1987 it is expected that the parliament will decide on a new act for radiation protection that will more clearly cover all areas of radiation protection e g protection of patients and protection of the environment.

Assignment of responsible parts

The radiation protection authority appointed by the government is the National Institute of Radiation Protection (NIRP) in Stockholm. It issues licences, gives regulations, makes inspections, measures occupational exposures and makes environmental measurements. So it has the double role of giving conditions and rules and supervise the observance of them. However, the main emphasis is on self-supervision i e each license-holder depending on the size of the establishment and the circumstances has to organize and perform an own appropriate supervision by education, measurements etc in combination with a system for reporting results, significant divergences from normal conditions, incidences, and accidents to the radiation protection authority. The authority gives general rules, makes sample tests, examines reports and makes irregular inspections. This casting of the parts implies a mutual confidence and has worked out very well in Sweden.

Radiation protection matters are mainly handled by one authority, NIRP, even though it cooperates with other authorities e g the Swedish Nuclear Power Inspectorate, the National Board of Health and Welfare, the National Environment Protection Board, the National Board of Occupational Safety and Health etc in the area of mutual interest. In the board of NIRP there are representatives of these authorities, employees' organizations as well as active politicians to get a broad social anchorage of decisions taken on policies and important affairs.

The principles of radiation protection

The principles of radiation protection in Sweden are based on ICRP's main recommendations, which means:

- Justification. No practice shall be adopted unless its introduction produces a positive net benefit

- Optimization. All exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account

- Dose limits. The dose equivalent to individuals shall not exceed the limits recommended for the appropriate circumstances by the Commission.

Assessment of the gross benefit can only be made in a broad social perspective. In certain cases these assessments are basically of political nature and they must be made at a higher level than radiation protection authority. This is the case for example in case of nuclear power. In other cases it is necessary to consult a governmental authority or other expert body with competence in the field of interest. When assessing the justification of medical exposure it is the responsibility of medical authority or the individual doctor. In case of research projects it is necesssary to consult the granting scientific council.

The justification process should include all negative and positive parts of a planned operation. This means e g that any possible waste disposal problem should be included. This has not always been the case in the past.

Optimization of radiation protection means that the protection should be improved below the limits until further improvements are no longer reasonable.

Optimization is the most important radiation protection principle. It should be used by authorities as well as users of radiation. However, a strict optimization procedure is not always so easily made. Therefore in practical radiation work it is often necessary to apply simple rules of thumb based on the authority's optimization deliberations.

There are various methods to optimize. A common method is called differential cost-benefit analysis which means a comparison of cost of protection and cost of detriment i e the radiation dose to man. This means that the detriment should be expressed in monetary units which presupposes political considerations. How much it is worth investing to prevent a particular addition to the radiation dose, is at the end determined by the total resources of the society and its level of ambition in the field of radiation protection. If the dose is expressed as collective dose in units of mansievert the monetary "value" of a mansievert as applied in various countries varies from about 1000 to 20 000 US$. What is paid in practice to save a mansievert varies in different occupations. It is lower in hospitals as regards protection of patients than in e g the nuclear industry, which in Sweden as in other countries has a very high ambition level of protection. In Sweden it is recommended that a value of 20 000 US$ per mansievert would be appropriate for optimization of radiation protection.

The limits used in Sweden are those given by ICRP e g 50 mSv per year for workers and 1 mSv per year for the public (5 mSv a^{-1} in a single year).

For the planning of the radiation protection in various practices that can irradiate the public by direct exposure from the practice, e g the use of consumer products, or indirectly by releases from the practice, e g releases from nuclear power, it is important that only a small fraction of the total limit of 1 mSv is used, e g 0.1 mSv from nuclear power. Doing so, exposures above 1 mSv because of addition of several exposures can be avoided. The real exposure of workers is often much lower than 50 mSv, of the order of 5 mSv per year. The irradiation of the public by releases of radioactive materials from e g nuclear power installations is also very small, a factor of more than 10-100 less than 1 mSv per year.

Practical application and problems

Medicine

The medical exposure in Sweden is the second highest exposure of people, about 1 mSv per year. The natural exposure is about four times higher. Because of the significance of medical exposure the priorities at hospitals are given to the protection of patients before protection of workers, the doses of which nevertheless are low.

Gynaecological radiology with radium needles still is a problem because of relatively high occupational exposures, particularly on hands. After-loading technique using other nuclides is pursued. In case of young patients with insertion or injection the parents are allowed to stay with the child that would result in doses to the parents of the order of a few millisievert.

Hospital physicists are responsible for the radiation protection of the workers and the medical doctor is responsible for the protection, examination and treatment of the patient although with due help of the physicist. New methods and nuclides in nuclear medicine are always first examined and approved by a special hospital committee composed of physicians, health physicists and pharmacists.

The justification of using radiation is usually judged by the physician. However, the justification of large examinations like mass-screening examinations using mammography are judged by central authorities.

Optimization in medicine means an adjustment between the medical outcome, the dose to the patient, the cost of treatment with and without use of radiation and the cost of protection. That is not an easy exercise and not merely optimization of protection. This can sometimes lead to the dose to the patient being increased e g to achieve a better quality of an x-ray picture, all for the benefit of the patient. Nevertheless by improving the technique the average dose per image has in general decreased in x-ray diagnostic examinations.

Research

The radiological work at research institutes is often characterized by great flexibility and variations. The radiation protection is sometimes based on the competence and judgement of the scientists, but this is not always so. The scientists may also underestimate or ignore the radiation protection problems. At large research institutions there are specially appointed groups and officers, who are responsible for the radiation protection. The justification of particular research work can hardly be made by a central radiation protection authority but will be decided by the granting scientific council or the responsible professor. The potential risk of internal contamination at many research institutions and laboratories has to be recognized and considered.

Industry (except nuclear power)

Four areas of applications can be mentioned:

1 Stationary and fixed apparatus with moderate activity (10^7 - 10^{10} Bq). In this case the radiological safety is inherent i e there are such technical requirements on source and equipment that no particular competence is needed to use the apparatus in a safe way. Normal exposures are reduced to a minimum by automatization, distance and shielding. Service and source-replacements are made by qualified experts normally from the contractor. The various types of apparatus are inspected at NIRP and the manufacturer is charged for the inspection. If the user wants to do his own service or there is a more complicated application there are special courses to improve the competence. Examples of apparatus mentioned are thickness-gauges and level-indicators with Kr-85 and Co-60 sources respectively.

2 Gamma radiography equipment with high activity (10^{11} - 10^{12} Bq). Some of these are stationary and used in a

special laboratory with thick shielding walls and an automatic fail-safe equipment. However, many of them are moved to various working places for gamma radiographic examinations and great requirements are made upon the competence and organization of the radiation protection work. The potential risk of accidents is great but fortunately no serious accident has yet occurred in Sweden even though there have been incidences and accidents with high overexposure of hands.

3 Gamma emitting sources for sterilization of medical instruments. Very high activity is used (10^{15} - 10^{16} Bq). The necessary extraordinary safety requirements means a high degree of automatization and there is no occupational exposure. NIRP makes special inspections and investigations at the time of source-change and other changes of the equipment.

4 Trace element investigation using low activity of shortlived nuclides like Na-24 for leakage investigations, tracing of industrial processes etc. There are requirements on very small concentration of impurities that can be activated in a reactor to other nuclides than the one of interest, that system for food production and for drinking water must not be affected and the general public should not otherwise be exposed by the investigation.

Consumer products

In Sweden the use of radioactive consumer products is restricted. The justification must be well verified, the activity should be low, the source and design should be inherently safe and there should not be any non-radioactive alternative (like batteries) that is competitive. On these conditions a large number of smoke-detectors containing small amounts of Am-241 are used in Sweden. After use they can be disposed of like inactive waste in the public garbage system. Other uses of radioactive consumer products are very limited.

Nuclear power

The 12 nuclear power reactors in Sweden are located at 4 sites, Forsmark 3, Ringhals 4, Barsebäck 2 and Simpevarp 3 all together about 9 GW_e installed. At each site there is special organization for radiation protection planning and control by the company itself. In principle this organization should be separated from the commercial and production organization. One or several named persons are responsible for the radiation protection at each reactor site and are also the contacts to NIRP. There are high requirements on their competence. General regulations are given by NIRP and the company has to give more detailed instructions for the normal daily operation. On a regular basis (month or quarter) the companies send reports to NIRP on measured occupational exposures and radioactive releases. In case of doses exceeding given values (10 mSv in a month) or release rates exceeding given values the company shall inform NIRP as soon as possible. By comparisons and tests the dosimetry and control systems of the company are checked occasionally.

NIRP has given a level of ambition for the occupational exposure expressed as the total annual collective dose per GW installed electric power which is 2 mansievert per GW_e installed. The real occupational exposure at Swedish nuclear power plants is very close to that level of ambition.

The company makes its own optimization of the radiation protection. However, if there is a special work where the expected collective dose exceeds 0.1 mansievert NIRP should be informed for discussion of the practical problems and possible improvements.

Optimization procedures have been used in the choice of systems for reduction of radioactive airborne releases. An optimum is found by differential cost benefit analysis to be a decay tank in combination with recombinators. The doses to the critical group of the public should be lower than 0.1 mSv a^{-1} per site irrespective of number of reactors. In practice the real doses are one to three orders of magnitude less. The releases into air and water are continuously measured.

The environment is controlled by sampling milk, plants, fish and water on a regular basis once per month to once per half a year. The samples are taken by NIRP or by radioecologists at the National Environment Protection Board. They are measured by the company and checks are made by the authorities.

At each community that has a nuclear power site, there is a safety board consisting of local politicians with the obligation and right to get information from the company and from supervising authorities about the safety and radiation protection matters of the plant.

The radioactive waste produced at nuclear power plants is taken care of by the companies concerned. Low active waste is burned or buried (shallow land burial) at the site in such small amounts and concentrations that the environmental impact from the burial is only a fraction of that caused by normal releases from the nuclear power reactors and no institutional control should be necessary after 100 years. The intermediate and high level waste will be buried deep in the rock, geological disposal. Some of the criteria for the radiation protection for this are the following:

- the assessment of radiological consequences should include all people irrespective of time and location

- they should include all doses irrespective of magnitude

- the efforts to avoid doses should be the same irrespective of when and where they occur

- the dose to critical group should be only a small fraction of 1 millisievert per year

- doses and consequences after 10 000 years are difficult to assess because of unknown environment and the assessment might have little value. However, a minimum requirement is that the doses and consequences would be acceptable assuming an unchanged environment

- a basis for judgement might be achieved by comparison with the flow of natural radioactive elements through the environment by weathering of the rock, transport by water into the biosphere and eventually ending in the sea.

The present emergency planning and preparedness is as follows:

- Mainly planned for nuclear accidents at Swedish nuclear power plants

- The emergency planning area around the nuclear power plant is divided into different zones with special emergency planning. The central alert zone extends to 5-10 km, the emergency zone 12-15 km and the radiation measurement zone to around 50 km from the plant

- In the radiation measurement zone there is preparedness for measurements by mobile instruments, information to farmers

- In the emergency zone furthermore there is (1) a network of permanent measurement sites with TLD, (2) iodine tablets in advance to each family, (3) an alert system for telephone alarm signals, (4) an information brochure to each family, (5) plans for evacuation

- In the central alert zone furthermore a permanent system with sirens for outdoors alarms

- The respective county administration authority is responsible for the local planning and actions taken in an accident situation. An advisory group at NIRP in Stockholm with specialists, to get information, collect data, calculate consequences and trends and give advice to the county administration authority

- Training and exercises on regular basis for the personnel concerned (e g a large exercise each 4 years for every site).

After the Chernobyl accident changes will be made to the emergency preparedness mainly in the way that all counties that have no nuclear power plant will also have some kind of preparedness to be able to handle long distance dispersion of accidental releases and subsequent ground contamination. Improvements of measurement systems, education and communication systems are being carried out.