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Note to the reader: *Since the creation of the* ***NEA Newsletter*** *in December 1983, one special and four regular issues have been published. Given the encouraging response, publication of the* ***Newsletter*** *will be continued on a regular twice-yearly basis. In an effort to improve presentation and make the* ***Newsletter*** *more attractive to readers, a new format which includes the ordering of sequence by volume and number has been adopted with this issue.*

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The OECD Nuclear Energy Agency (NEA) was established in 1972, replacing the European Nuclear Energy Agency. The NEA groups the 19 European Member countries of OECD together with Australia, Canada, Japan and the United States. The Commission of the European Communities and the International Atomic Energy Agency take part in the NEA's work.

The purpose of the NEA is to further the development of the peaceful uses of nuclear energy by sponsoring economic, technical and scientific studies and projects, and by contributing to the optimisation of safety and regulatory policies and practices.

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The continued suspension of sea dumping of radioactive waste

J.-P. Olivier

The production of nuclear power, like any other industrial activity, generates wastes which require proper handling, conditioning, storage and disposal. The management of radioactive waste continues to attract considerable public attention, and national authorities are often faced with the challenge of having to reconcile scientific and political viewpoints.

One example of this is the disposal of low-level radioactive waste. There are two basic options available, both of which have been in use up to now. One is the emplacement of this waste in shallow-land burial facilties or in deeper underground structures. The other option consists of disposing of the wastes, conditioned in appropriate containers, at an approved site in the deep ocean.

Sea disposal activities are strictly regulated by the Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter (the London Dumping Convention, LDC), as well as by technical requirements and mechanisms established by the International Atomic Energy Agency (IAEA) and the OECD Nuclear Energy Agency (NEA).

The LDC Decision

At the Ninth Consultative Meeting of Contracting Parties to the LDC, recently held in London, a resolution calling for the suspension of all dumping at sea of packaged radioactive wastes, pending the completion of a number of studies specified in the resolution, was adopted by a majority vote. A similar resolution had already been adopted at the Seventh LDC Meeting, in February, 1983, together with a request for further scientific advice on the acceptability of the sea dumping of radioactive waste. This scientific assessment was prepared by an independant Panel of Experts nominated by the International Atomic Energy Agency (IAEA) and the International Council of Scientific Unions (ICSU). It was further discussed in June, 1985 within an Expanded Panel which included experts designated by the Contracting Parties. The conclusions of the scientific assessment were made available at the Ninth Consultative Meeting of the LDC as a basis for the Convention to decide whether sea dumping of packaged radioactive waste, which was practiced by a number of countries until 1982, could continue to be used as a safe disposal option.

The Expert Panel assessment

The problem confronted by the experts was that such assessments have to rely exclusively on a scientific modelling approach because:

- The radiation doses resulting from radioactive waste disposal into the sea are too small to be determined by direct measurements;
- Radiation doses occur in the future, after the dumping has taken place.

The modelling approach requires an understanding of the many physical, chemical and biological processes contributing to the dispersion of radioactive materials in the marine environment and their potential effects on man and marine organisms. It is also necessary to know specific data to determine the parameters of the models, such as oceanographic data and concentration factors of radioactive material along the marine food chains.

The Panel of Experts had at its disposal a large body of scientific and technical information, both in the radiological protection field and in the area of marine radioecology, where research programmes have been conducted since the beginning of nuclear energy activities. Included in this was the recent NEA review of the continued suitability of the North-East Atlantic radioactive waste dump site, which has been used by a number of European countries over the last ten years. This review of the only existing disposal site is the most extensive and up-to-date radiological assessment done on this subject. It draws extensively on the results of the NEA Co-ordinated Research and Environmental Surveillance Programme (CRESP), carried out since 1981 by ten countries with the co-operation of the IAEA.

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On the basis of the available information, the Panel of Experts considered that they could make an assessment which, in its estimates of exposures to man and marine organisms and in its interpretation of these results in terms of radiation protection, was sufficiently reliable to allow the conclusion that: "No scientific or technical grounds can be found to treat the option of sea dumping differently from other available options when applying internationally accepted principles of radioprotection to radioactive waste disposal".

These principles of radiation protection, which are recommended by the International Commission on Radiological Protection (ICRP), require compliance with individual dose limits as well as the use of an optimisation approach (*i.e.,* comparing the available disposal options with the ultimate choice of a solution that keeps all doses as low as is reasonably achievable, social and economic considerations being taken into account). In practice, this involves a series of assessments and decisions:

- First, it is imperative to distinguish among disposal options those which are clearly unacceptable from the point of view of individual doses or risks and those which *might be* acceptable; in essence this was the mandate given to the Expert Panel.
- Second, collective dose considerations and comparison between alternative options have to be factored into the optimisation of protection.
- Third, non-radiological protection factors need to be taken into account from a more global perspective, a step outside the terms of reference of the Expert Panel.

From the point of view of the experts, sea dumping of radioactive waste could not be ruled out solely on the basis of generic radiological protection considerations. The individual doses estimated for both man and the marine biota are sufficiently low to make the practice of sea dumping acceptable in principle. This conclusion had already been reached more than a decade ago and had served as a basis for setting up the frame of the LDC, which was supplemented by the IAEA technical requirements. The experts were not, however, in a position to make a meaningful comparison with other land-based disposal alternatives. This would have required an assessment of actual cases in a specific context, an assessment which can usually be done only by national authorities who would have all the necessary information. The experts recognised the need to proceed with such comparisons and the optimisation step before dumping permits could be granted, according to IAEA Recommendations.

The LDC review and its conclusions

The Expanded Panel of Experts concurred in large part with the findings of the original Panel. It was confirmed that improvements could be expected in the future, but given the analysis of the sensitivity of models to variations of environmental parameters, the

How the London Convention works

The 1972 London Convention lays down strict rules for the dumping of all types of pollutants, both chemical and radioactive. They prohibit the dumping of a whole range of substances, including highly radioactive waste. The dumping of other substances is allowed, subject to the prior issue of special or general permits by the responsible national authorities who must comply with the criteria and conditions laid down by the Convention. A special permit is required to dump radioactive waste that is not prohibited. In issuing such permits, the authorities must take account of the recommendations of the International Atomic Energy Agency (IAEA), the body which is given responsibility by the London Convention for defining highly active wastes – *i.e.,* those that are to be considered unsuitable for dumping at sea – and for formulating recommendations concerning the conditions in which dumping of the other radioactive wastes may take place. The definition and recommendations were drawn up in 1974 and have been periodically revised. (The latest version was adopted by the IAEA Board of Governors in September 1985.) The recommendations provide for detailed ecological and environmental assessments prior to dumping and set forth requirements for selection of dumping sites, for conditioning and packaging wastes and for the ships themselves, and also for supervision of operations by escorting officers on board.

The OECD Consultation and Surveillance Mechanism for Sea Dumping of Radioactive Waste was set up in 1977 to further the objectives of the London Convention. The Mechanism provides for:

i) The establishment of standards, guidelines and recommendations concerning the scientific, technical, environmental and operational aspects of sea dumping operations;
ii) Prior consultation amongst participating countries on the detailed conditions proposed by national authorities for any given dumping operation, to ascertain whether it conforms to established rules;
iii) International surveillance of dumping operations by specifically appointed NEA Representatives carrying out their duties in co-operation with the national escorting officer;
iv) International examination of the details of the execution of operations and recommended improvements as appropriate.

most important results appear to be relatively insensitive to such variations. For the marine biota, it was again confirmed that there is no likelihood of any significant damage. Individual doses to man and the associated risks are very low indeed, whatever standards of comparison are chosen.

The need to provide a comparison with land-based alternatives and the interpretation of collective dose assessments was again discussed. As already explained, such comparisons need to be realistic and site-specific for both sea dumping and land-based alternatives. Therefore, they are best conducted at the national level. However, even if a comprehensive environmental assessment is made at the national level in the context of a proposal to carry out a sea-dumping operation, the conclusions of these comparative studies will be reviewed internationally.

For a majority of countries at the Ninth Consultative Meeting, the scientific background was only part of the considerations to be taken into account in judging the overall acceptability of the sea dumping of radioactive waste. The importance of social and moral factors was stressed, as well as the question of "burden of proof", requiring the demonstration of "absolute safety" for such operations. On these issues the Consultative Meeting was divided among countries opposed to sea dumping as a matter of principle or for other reasons, those in favour of a continuation of the present special permit regime of the LDC, by which sea dumping of radioactive waste is allowed under specific circumstances, and those who preferred not to take any position, primarily because of difficulties in reconciling political opinion with the conclusions of a scientific assessment.

Although efforts were made to reach a consensus and avoid a split between Contracting Parties which might affect the credibility of the London Dumping Convention, a vote was requested by a large majority of countries. As already mentioned, the resolution which was adopted called for a further suspension of the dumping of radioactive waste, pending the conclusions of a number of studies and assessments. It was noted, however, that the scope of some of the further studies mentioned in the resolution may need to be better defined and that this could be discussed at the Tenth Consultative Meeting of the LDC in October, 1986. These studies include, among other things, consideration of political, legal, economic and social aspects of radioactive waste dumping, comparison with land-based options, the question of the burden of proof, and additional scientific work by the IAEA.

The effects of the resolution

The resolution, which was adopted by a simple majority*, is not legally binding, as the Convention itself and its Annexes have not been modified; this would have required a two-thirds majority. Nevertheless, such a resolution, given its political significance, will make it difficult for any country to proceed with a dumping operation in the near future. Since doubts were expressed at the meeting about the feasibility of carrying out some of the studies listed in the resolution, it is likely that the open-ended suspension will be maintained for a relatively long period of time.

On the other hand, the countries which have proceeded recently with sea dumping operations or which might wish to carry out sea dumping at some point in the future, have all declared that they have no specific plans for the time being. Most of them are presently looking for land-based alternatives. Some countries are in the process of conducting a broad comparison of all available alternatives and the conclusions of these studies are expected soon. It cannot be excluded that, in specific situations, sea dumping will appear as the preferred option.

More generally, two remarks can be made about the significance of the LDC resolution. First, to a large extent the scientific background has been disregarded. In addition to risking the credibility of the LDC itself, this might have a bearing on research programmes and the interest of scientists in an area where safety and environmental assessments are probably the most advanced and could be used as models for the control of pollution from other sources. Second, from a general radioactive waste management standpoint, the LDC resolution suggests that present political constraints may prevent *a priori* the use of international solutions for the disposal of radioactive waste, however safe they might be. This means that temporary or alternative solutions at the national level have to be relied upon which may, in an optimisation approach, prove to be less satisfactory from the standpoint of both risk protection and utilisation of resources ■

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* *In favour:* Australia, Brazil, Chile, Cuba, Denmark, Dominican Republic, Finland, Federal Republic of Germany, Haiti, Honduras, Iceland, Ireland, Kiribati, Mexico, Nauru, Netherlands, New Zealand, Norway, Oman, Panama, Papua New Guinea, Philippines, Saint Lucia, Spain, Sweden. *Against:* Canada, France, South Africa, Switzerland, United Kingdom, United States. *Abstentions:* Argentina, Belgium, Greece, Italy, Japan, Portugal, USSR.

Source terms: Evaluating new information

D.F. Torgerson

One of the most intensive areas of nuclear safety research over the past 10 years has concerned "source terms". In general, "source terms" characterise the potential release of radioactivity following a severe reactor accident. A severe reactor accident is an accident in which the core of a reactor is sufficiently damaged (due to loss of cooling) so that fission products are released from the fuel matrix. Such releases include those associated with fuel melting and with pressure vessel (the vessel that contains the core) melt-through. It is important to note that source terms may be defined differently depending upon the end use of the information. For example, to an analyst assessing the performance of a reactor containment building, the source term describes the radioactive material released from the reactor core to the containment building. To an air cleaning specialist, the source term describes the challenge to a filtration system. However, from a nuclear safety point of view, the most widely used definition is that the source term is the quantity, timing, and characteristics of the release of radioactivity to the environment following a postulated severe accident.

Source terms are used by regulators for such activities as emergency planning, risk assessment, setting research priorities, evaluation of potential backfits, and the resolution of safety issues. Obviously, any changes to source terms could have significant impact on these regulatory activities, and on the utilities operating nuclear power plants. Over the past few years, there has been considerable progress in the development of our knowledge of source terms. This article summarises the current situation, and indicates how the NEA's Committee on the Safety of Nuclear Installations (CSNI) is evaluating the new information.

Barriers to release of radioactivity

Although severe accidents have low probabilities of occurring, the calculated consequences of such accidents could be high if source term values associated with severe accidents are large. For this reason, it is useful to review the principal barriers to the release of radioactivity, assuming that the various engineered safety systems are not operating. The first barrier is the fuel, which is a dense uranium oxide (UO_2) matrix in which most of the fission products are fixed. The UO_2 matrix is surrounded by a metal sheath, such as Zircaloy or stainless steel. If the fuel becomes sufficiently hot, the sheath fails and relatively small amounts of fission products are released into the primary coolant circuit – *i.e.,* the "gap inventory" of fission products that are between the sheath and fuel matrix. If the temperature continues to rise, for example to melting temperatures, then more fission products would be released from the fuel matrix.

The next barrier is the primary circuit, which contains the core and the water coolant. Fission products may be depleted in the primary circuit by a number of mechanisms, depending on the particular accident sequence. In very severe accidents, the core may melt through the primary vessel into the containment building, and additional radioactivity would be released from the fuel, due to the interaction between the melted core and the concrete basemat.

The containment building surrounds the reactor and is designed to withstand high pressures and temperatures. Within the containment building, natural processes occur that deplete radioactivity from the gas phase. In addition, there are other removal processes due to the operation of devices such as water sprays and fan coolers. As we shall see later, a key ingredient contributing to improved source term values is the progress that has been made in understanding performance of the containment building during a severe accident.

The 1975 WASH-1400 Reactor Safety Study

The modern history of the source term begins with the WASH-1400 Reactor Safety Study, commissioned by the US Atomic Energy Commission, and published in 1975. The study classified the source term values into "release categories" that were associated with various levels of damage to nuclear power plants.

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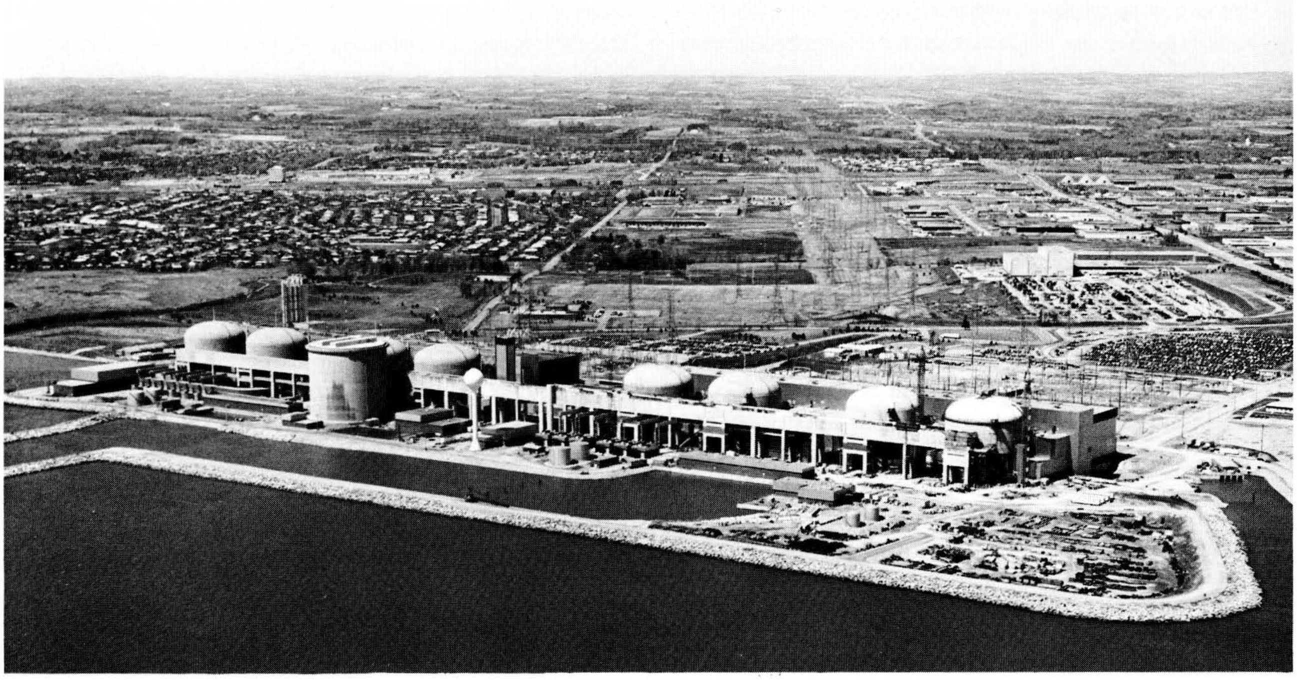


The Pickering "A" and "B" Generating Stations in Canada
Credit: **Ontario Hydro**

At the time of WASH-1400 it was recognised that there was little knowledge of some of the important phenomena associated with source term technology. In some cases, the important phenomena had to be neglected (such as the effectiveness of some engineered safety systems), and in other cases, simplified models had to be used that contained conservative (*i.e.,* pessimistic) assumptions concerning the behaviour of radioactive material. In particular, such phenomena as fission product retention in the primary coolant circuit, steam condensation in containment buildings, aerosol behaviour in containment buildings, fission product release pathways, and some important aspects of the chemistry of volatile fission products were largely neglected. The result was that many of the source terms predicted in WASH-1400 were highly pessimistic with respect to the quantity and timing of the release of radioactivity from nuclear power plants. However, WASH-1400 was the only comprehensive description of reactor accidents available to regulators, and the information in WASH-1400 is still used today in several areas of regulation.

New information

Since WASH-1400, there has been considerable research activity that has led to a better understanding of the source term phenomena that were originally neglected. As a result, a number of new studies have recently been prepared to reassess source term technology. These include reports prepared by the American Nuclear Society, the American Physical Society, the Industry Degraded Core Rulemaking Program, and work performed by, and for the US Nuclear Regulatory Commission. Various other studies have been done by such organisations as the Stone and Webster Engineering Corporation, the Electric Power Research Institute, and the New York Power Authority. More specialised reports have also been prepared, based on work in several NEA Member countries, including Canada, France, the Federal Republic of Germany, Italy, Japan, Sweden, the United Kingdom, and the United States. Finally, a vast pool of relevant technical information has become available from research programmes in various countries.

At the November, 1984 meeting of the NEA's Committee on the Safety of Nuclear Installations it was decided to organise a Special Task Force on Source Terms to review the studies as well as ongoing work. Since most of the studies were available by early 1985, the Task Force began its work in February and completed the review in October, 1985. The Task Force consisted of an international group of scientists and engineers who are experts in the various technical areas of importance for evaluating source terms. A few examples of the Task Force's findings will serve to illustrate the impact of the new information on source terms.

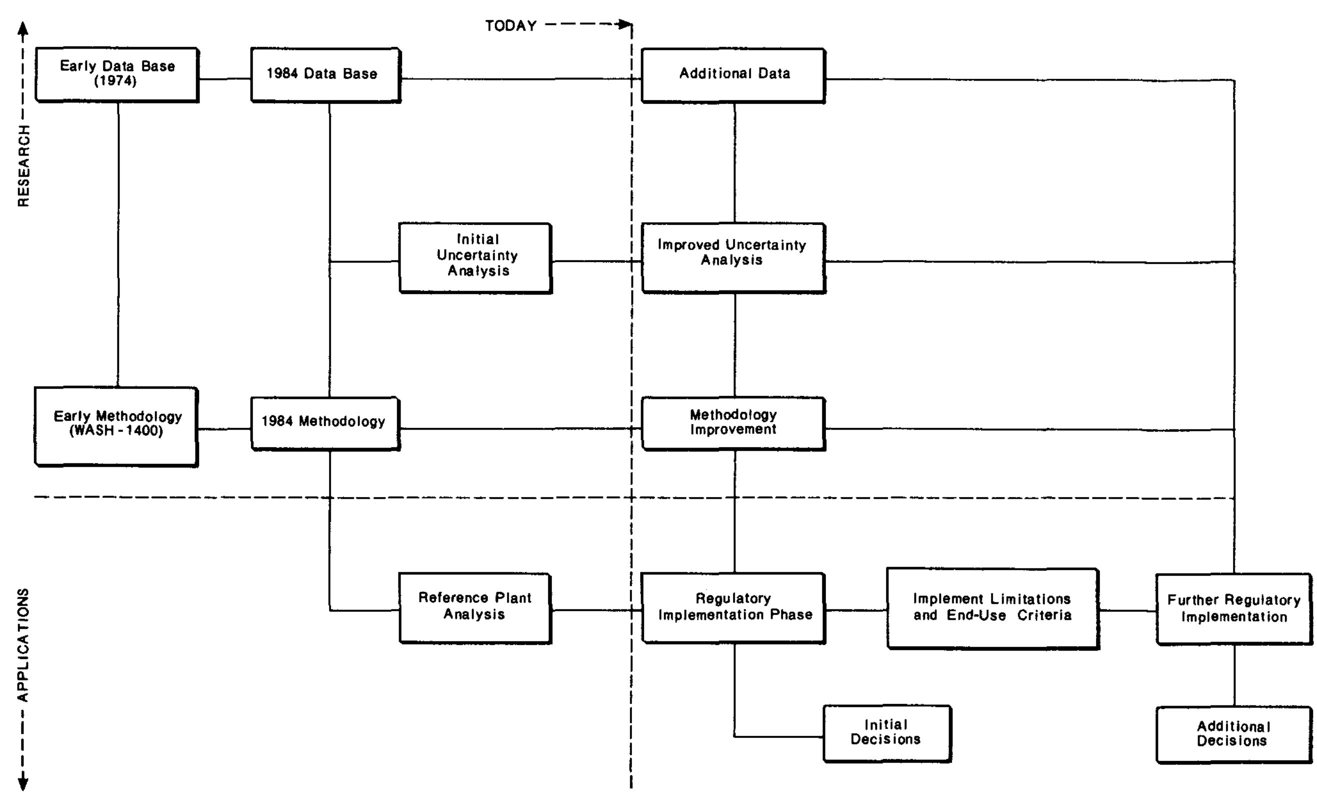
One of the main developments since the 1975 WASH-1400 report is that most source terms for nuclear power plants can now be calculated on a mechanistic basis. That is, the important mechanisms have been identified and source term values can be based on technical information, not on pessimistic assumptions. As a result of this, many of the old source term values have been found to be over-estimated, sometimes by large factors.

Another major factor resulting in reduced source terms is a better understanding of the performance of nuclear power plants. For example, some containment buildings have been found to be 2-4 times stronger than their design pressure. During a severe accident, these containment buildings would fail, if at all, after longer times. This is important since recent information on fission product/aerosol behaviour in containment buildings during a severe accident indicates that most of the radioactivity would not be in the gas phase at the longer times, and therefore would not be available for release.

There have also been significant advances in the understanding of fission product chemistry during reactor accidents. For example, at the time of the WASH-1400 study, it was known that radioiodine would likely react with cesium (like iodine, cesium is a fission product formed in fuel, but is about 10 times more abundant than iodine) to form the low-volatile, water-soluble salt, CsI. However, since there was a poor data base for characterising this reaction, it was assumed that iodine would form volatile iodine (I_2). This led to very large source terms for iodine release in some postulated accidents. However, the importance of CsI formation became strongly evident during the Three Mile Island accident in 1979. Although a large fraction of the core inventory of iodine was released from the reactor core, the iodine source term was only a very small fraction of what had been expected if the iodine had formed I_2. Today, there is sufficient information to characterise CsI formation during reactor accidents and in many accidents much lower iodine source terms are justified.

The dependence of fission product transport on thermalhydraulics (the flow of heat and mass) in reactors is another area where there have been notable advances. Most of the important in-vessel thermalhydraulic phenomena are now recognised, and the current methodology is probably adequate to predict fission product retention in the primary coolant circuit. Also, the improved understanding of aerosol transport in containment buildings is allowing analysts to couple aerosol physics with containment thermalhydraulics.

Figure 1. **RELATIONSHIP BETWEEN SOURCE TERM RESEARCH, UNCERTAINTIES AND IMPLEMENTATION**



The above are just a few examples of new information and how the new information is affecting source term values. At the same time, the new information has resulted in the definition of phenomena that need further characterisation. Examples are the need to improve the models for core-slump behaviour, for core concrete interactions, and for some aspects of hydrogen combustion. These areas are all being addressed in international research programs, and it is highly unlikely that any important phenomena are being neglected.

Remaining uncertainties

Although source terms for many accident sequences have been over-estimated in the past, it may prove difficult to arrive quickly at specific source term reduction factors that are universally acceptable. The main reason for this is that different code sets may result in different source terms, depending on the specific plant and accident sequence being analysed. These differences can be traced to differences in the mathematical models representing the physical phenomena, differences in the physical properties of materials, possible omission of important phenomena, specifications of the accident sequence and plant geometry, and numerical approximations.

One way of characterising uncertainties is to recognise that the importance of a particular uncertainty depends on the timing and mode of containment failure. As discussed previously, stronger containments reduce the probability of early containment failure, and aerosol depletion processes reduce the impact of late containment failure. Thus, for these stronger containment buildings, uncertainties associated with early and late containment failure would have less impact than the uncertainties associated with intermediate times. The reduction of these remaining uncertainties will undoubtedly receive the highest priority in future research programs.

Regulatory implications

The remaining uncertainties notwithstanding, all the current studies indicate that the source terms for many postulated accidents can be reduced from WASH-1400 values. The implications of the current work for regulatory activities are summarised in Figure 1, which shows the historical, current and future relationships between source term research and eventual applications. The early 1974 data base has now been replaced by the 1984 data base, which includes all the information available at the time of preparation of the recent studies.

While there is always a temptation to delay implementation of new information until the last "i" is dotted and "t" is crossed, most source term experts feel that efforts should now begin to use the new information in regulatory applications. The vertical dashed line in Figure 1 is the current situation. The additional data, improved uncertainty analysis and improved methodology arising from research activities should now lead to a regulatory implementation phase. The remaining research to be done should be part of an interactive process, whereby comparison of the application requirements (as developed by regulatory agencies) with the remaining source term uncertainties will determine the need for future research or code improvements. Such an approach is feasible today owing to the outstanding progress that has been made in source term technology ■

Decommissioning large power reactors: Strategies

K. Bragg

Over the next five years, eleven moderate-size reactors within the OECD area (where there are some 260 existing reactors) will need to be decommissioned. The term "decommissioning" covers all operations conducted on a reactor once it has ceased generating power and after its last fuel charge has been removed. Since the largest amount of radioactivity is contained in the spent fuel, this latter stage is extremely important. The time between actual shut-down and the beginning of decommissioning is typically a few years and several large reactors in the OECD area have already undergone defuelling.

Decommissioning is not a single operation or small series of operations but rather a complex series of inter-related steps. The International Atomic Energy Agency (IAEA) defines decommissioning as having three stages.* The first stage consists of removing or recycling some of the support buildings and equipment which are either radiologically uncontaminated or lightly contaminated. The more radioactive components directly connected to the reactor can then decay under an active surveillance regime. This stage might include regular inspections to ensure that there are no leaks in the containment building and that ventilation equipment and other services required to keep the installation in safe condition are available as needed. The second stage consists of removing most of the equipment and buildings up to the reactor core and its heavy biological shielding. The remaining components contain most of the residual radioactivity and are conditioned in a way which requires less active surveillance than Stage 1. This holding period allows short-lived isotopes to significantly decay and may extend for a period of 50-100 years. The third, or final, stage removes all components and returns the site to a green field state. Contaminated materials are brought to a licensed disposal site. Decontaminated buildings or equipment could be used for any new purpose. The site of the reactor itself would be completely decontaminated and also suitable for any type of use.

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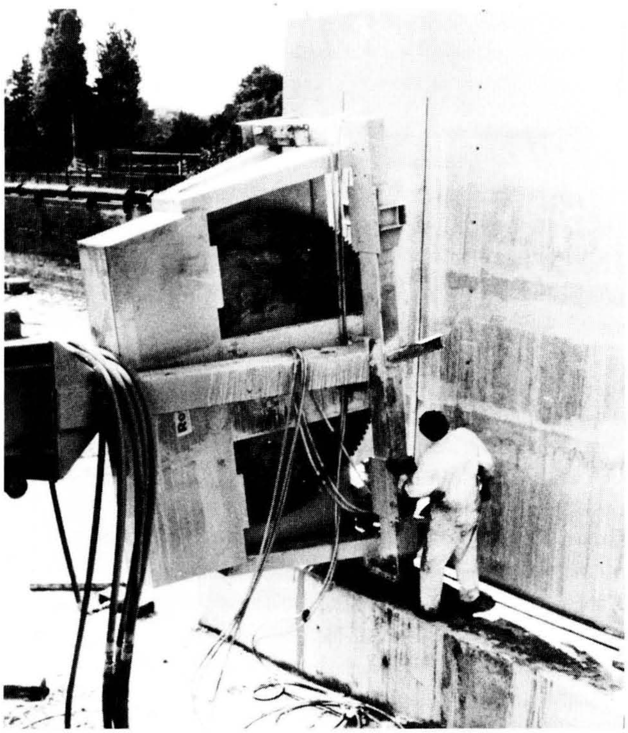
* IAEA-TEC-DOC 179 *Decommissioning of Nuclear Facilities*, 1975.

It must be stressed that Stage 3 will be carried out in all cases and that Stages 1 and 2 are merely interim steps to facilitate that end. The major choice to be made is the sequence of steps. Conceptually, the simplest step is to dismantle immediately (*i.e.*, go straight to Stage 3). However, there are technical, safety and economic factors which may make a slower approach more attractive. Thus, one can undertake either Stage 1 or Stage 2 initially, and follow it by Stage 3. This paper will attempt to outline the main factors which may affect this choice.

Technology

A critical element in choosing among the stages is the availability of technology to undertake the necessary operations. In other words, does the technology exist today or does one need to wait for future developments, thus eliminating the immediate Stage 3 option? To understand this matter, it is necessary to examine the kinds of work which must be carried out. These fall into broad categories, such as decontamination, dismantling and waste management. Decontamination involves removing radioactivity which has deposited on the surfaces of components. Since it is attached to surfaces, it is possible to remove a large proportion using physical cleaning (scraping, stream cleaning and sand blasting) and chemical methods such as strong acid dissolution and electropolishing. The small amounts which cannot be removed in this way are very firmly attached to the pieces and thus are not easily mobilised and cannot readily enter the environment.

Another type of contamination is due to neutron activation of components near the reactor core, during normal operation. This radioactivity exists within the bulk of the material and thus is not amenable to classical means of surface decontamination. The approach is therefore to dismantle such equipment and handle it as radioactive waste in licensed facilities. The methods used for dismantling include various types of cutting such as shearing, arc and plasma torches and diamond-edged saws. Explosives are also used to help in dismantling bulky structures.



Cutting Saw Developed to Saw Reinforced Concrete at the Windscale Advanced Gas Cooled Reactor Decommissioning Project

Finally, the wastes arising from dismantling and decontamination need to be managed in appropriate ways. The wastes are segregated into various classes to be disposed of at different types of facilities, such as shallow land burial sites and deep geological repositories. They must also be packaged into forms compatible with the physical and chemical properties of the waste and of the waste disposal facility.

The key technological issue is, therefore, whether all of this can be done while effectively protecting the workers, the public and the environment and at a reasonable cost. A recent NEA workshop to compare immediate and delayed dismantling concluded clearly that it can be done.* This is reinforced by the work (still in progress) of an NEA Expert Group on the technical and economic feasibility of decommissioning. In effect, all the necessary operations have been tested either on a full scale or a pilot scale. The experts have agreed that full application to large power reactors is possible and several large demonstrations are planned or are underway.

To help develop the operational experience needed for the actual decommissioning of large reactors, the NEA is setting up an international co-operative programme whereby exchange of scientific and technical information will be organised around a number of national decommissioning projects which are of comparable magnitude or degree of technical interest.

* NEA, *Proceedings of the NEA Workshop on Storage with Surveillance vs. Immediate Decommissioning For Nuclear Reactor Components and Buildings,* 1985.

Safety

As mentioned earlier, the largest portion of the total radioactivity in a reactor is in the fuel elements and these are removed for storage or disposal prior to the start of decommissioning. The next largest portion is due to activation of the reactor core. Here more than 90 per cent of the residual radioactivity is contained in less than 10 per cent of the waste volume. The decommissioned reactor is thus a much smaller source of potential exposures to the public than the operating reactor and this has been confirmed by actual monitoring around decommissioned reactors. Worker protection, however, is a key element and to a large extent determines what decommissioning operations are to be carried out and how they should be done. Direct experience has shown that it is possible to protect workers using conventional techniques like shielding and area monitoring along with remote handling equipment. Worker protection may be further facilitated by allowing for radioactive decay, but this tends to be related more to cost considerations, since protection can be assured in all situations.

Costs

Another important factor affecting decisions on decommissioning is cost. There is an impression in some circles that these costs may be dominant in the entire nuclear power cycle, but this is not supported by recent studies. The preliminary results of the previously mentioned Expert Group on the technical and economic feasibility of decommissioning suggest that the total cost of decommissioning to Stage 3 will be a few per cent of the total cost of generating power over the life of the plant. The actual figures will, of course, vary with plant type and with the choice of decommissioning sequence and operations. Furthermore, considerable scope exists to optimise operations and to achieve increased cost effectiveness and there is an economic justification to continue research and development programmes. These will likely focus on waste volume reduction and packaging, methods to more accurately screen wastes, development of more robotic devices and systems and sophisticated decontamination and dismantling technologies.

Radioactive waste management

Wastes are generated at all stages of decommissioning and the general methods to handle them have already been discussed. Waste management, however, has a broader role to play in decommissioning decisions. For example, the criteria set by authorities can have a considerable impact on waste volumes. Thus, if some very low-level wastes are exempt from regulatory control, this may reduce costs without any adverse environmental effects. If Stage 3 (complete dismantling) is delayed for 50-100 years, the short-lived radionuclides will decay and therefore wastes may shift into categories which can be handled more easily, and more waste may also become exempt.

Therefore, even though wastes of all types can be safely and economically managed at any time, there is a benefit to be had from a delay. The particular benefit will be related to the reactor type, and to the spectrum of isotopes present in each waste type as well as the operating history of the reactor. Offsetting this benefit, however, is the need to provide for continued maintenance and surveillance throughout the deferral period.

Today, one aspect of waste management dominates decommissioning decisions: the availability of disposal facilities and the difficulty in finding waste disposal sites which are acceptable on social and political grounds. A recent NEA document* states that from a technical standpoint, suitable sites can be found within all countries with a nuclear programme. Where waste disposal facilities do not exist, it is clear that Stage 3 will have to be delayed.

Future decisions

Large power reactors can thus be safely decommissioned to any stage, either now or in the future, using existing equipment and techniques and at a cost which is only a fraction of the total cost of generating nuclear power. The volume of waste arising from decommissioning is such that it will not require a significant expansion of existing or planned disposal sites. It can also be managed using conventional techniques, provided that sites and facilities exist to receive them. The final choice of decommissioning strategy and the timing of implementation cannot be predetermined but depends on a complex balance of technical, safety, economic, political and social factors ■

* NEA, *Technical Appraisal of the Current Situation in the Field of Radioactive Waste Management*, 1985.

Smaller-sized power reactors: The market potential in OECD countries

H.E. Thexton

Early in the history of commercial nuclear power, economy of scale considerations led many of the large industrialised countries to build big plants, typically 900 to 1 300 MWe in capacity. However, since almost the same time, designers have considered means of building economic plants in smaller sizes more suitable for many developing countries with small electricity distribution systems. In 1983, the International Atomic Energy Agency (IAEA) initiated a comprehensive study to survey, amongst other things, available small to medium-sized power reactor (SMPR) designs and the potential market in developing countries for plants of this size range. Since early discussions indicated that industrialised countries were also interested in smaller plants, the NEA undertook to study the potential market in OECD countries and to provide a chapter for the IAEA report*. The NEA study, done with the assistance of a consultant, considered SMPRs and MPRs (medium-sized power reactors) covering the range of 200 to 700 MWe.

The study evaluated OECD countries, or in some cases regions within countries, which had an electricity distribution grid of a size appropriate to smaller reactors. The criterion used was that the reactor capacity should not be greater than 10 per cent of the grid capacity. Thus, systems with grids of 2 000 MWe to 7 000 MWe were considered. Of course, larger grids could also accept 200 to 700 MWe reactors, but in this case the inclination would be to choose larger units.

Another factor considered was the expected growth rate of the country's or region's demand. Here the criterion tested was that the unit size should supply two to five years expected increase in demand. If the growth rate was much faster than this, a utility would likely opt for a larger unit size rather than having to construct several units at almost the same time. If the rate was much slower, the system would have unused capacity available for several years after the unit commenced operation, and this would likely be uneconomic.

A third factor considered was a country's ability to finance a station. It was assumed that a country would not want to spend more than 1 per cent of its gross domestic product (GDP) annually to finance the capital charges of any one project. Since an SMPR is expected to cost on the order of US$ 1 billion, resulting in financing charges of about US$ 100 million per year, a country would have to have a GDP of at least US$ 10 billion per year to practicably finance one unit.

No OECD countries were found to fall within all of the above criteria. However, Greece, Ireland, New Zealand and Portugal only slightly exceed the two-year growth rate criteria and may be considered, technically, as possible markets for the 200-700 MWe size of unit. Several Australian states and Canadian provinces also meet all of the criteria, as do the Italian island of Sardinia, Japan's Hokkaido, and the US states and territories of Alaska, Hawaii (Oahu Island) and Puerto Rico.

The study emphasized that, although these countries or regions may meet the technical criteria for SMPR or MPR units, they may well find that fossil or hydraulic units are more appropriate for their applications. Social and political factors were not evaluated in the study, but it was noted that of the countries identified above, Australia, Greece, Ireland and New Zealand have adopted electricity planning policies which currently exclude nuclear power.

The report also considered markets in larger OECD countries and noted that there was a possibility that 200-700 MWe-sized units (particularly the upper-end of this range) might be considered under special circumstances. For example, some opportunities may exist for dedicated power supply for electricity-intensive industries (aluminum smelters, electrolytic processes, etc.) or for process heat supply (district heating, desalination, upgrading coal, etc.). Also, some supplier countries could choose to build demonstration units to further the prospects of export sales.

Certainly there is a significant market in OECD countries for plants in the size range studied. During the 1980s some 113 plants in the 200-400 MWe size range plus 164 in the 400-700 MWe range have entered service or are scheduled to do so. Most of these are fossil-fired units, many designed for peak load applications. Still,

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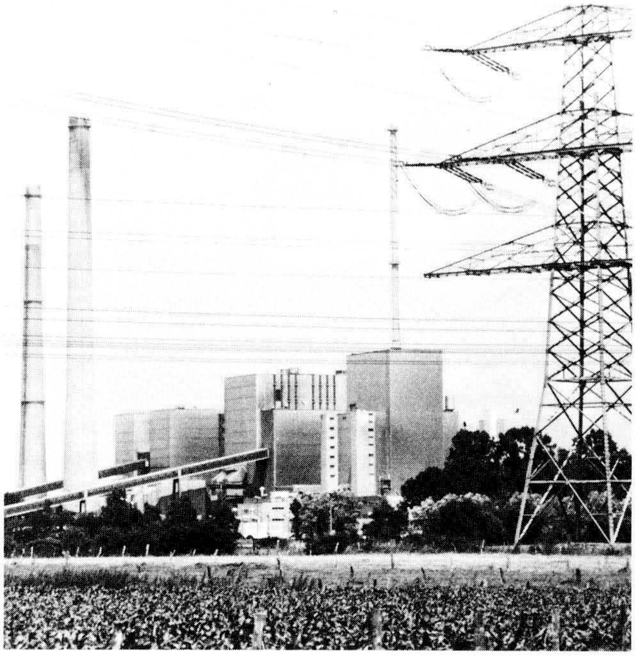
* IAEA-TECDOC-347 *Small and Medium Power Reactors: Project Initiation Study Phase I*, July 1985.



The Hamm-Uentrop 330 MWe High-temperature Reactor in the Federal Republic of Germany (built alongside coal plant at left)
Credit: **HRB**

many are base load plants and a few in the upper-end of the range are nuclear plants. Although nuclear plants in this size range have been relatively expensive in specific capital cost terms, making them less competitive with fossil-fired units, designers of proposed new smaller reactors are incorporating novel approaches to significantly reduce construction time and therefore financing costs. Building a series of small units on one site, to a standardized design and with schedules phased to optimise component fabrication and site construction, can further reduce costs. Building units which more closely match the rate of demand growth can also lead to system cost savings. Adding smaller increments can facilitate financing and provide somewhat greater flexibility in responding to uncertain rates of demand growth. Thus smaller reactors may not be as expensive as generally assumed on a total system generating cost basis, and they may well be competitive in some locations and under some circumstances. However, cost data is not yet known well enough to make generalisations; economics will have to be determined for particular cases on the basis of specific bids from suppliers.

Some 17 suppliers in seven countries have indicated an interest in supplying smaller reactors. Their main target markets will be the developing countries but they will certainly also be exploring opportunities in the OECD area. While large units seem certain to continue to dominate this market, some sales of smaller reactors appear to be quite probable ■

Update on the OECD LOFT project

R.R. Landry

The OECD Loss-of-Fluid-Test (LOFT) Project provides thermal-hydraulic fuel and fission product information used to assess computer codes, define safety margins, define previously unanticipated phenomena and develop techniques for accident recovery. The Project uses a pressurised water reactor located at the Idaho National Engineering Laboratory in the United States. * The OECD LOFT Project has been in full operation for more than two years and of the eight tests planned, six thermal-hydraulic tests have been performed. The last two tests, performed in December 1984 and July 1985, were the fission product tests. These reflect a significant change in the test programme and have far reaching implications for nuclear safety.

LOFT today

The experimental part of the LOFT programme has been successfully completed and the analyses of the results and the assessment of what has been learned must now be performed. The first six tests provided a wealth of information on the thermal-hydraulic response of a nuclear power system to an upset or accident condition. The last two tests have provided data that will help in understanding what happens when the fuel rods in a nuclear reactor undergo damage ranging from minor levels to melting. These two sets of tests were distinctly different in objectives and method of operation. Whereas the thermal-hydraulic tests used standard operating procedures and safety systems to investigate PWR response to events up to, and including, a full severance of a coolant pipe, the fission product tests deliberately inhibit the safety systems to permit the fuel rods to be stressed beyond their design limits and thus be damaged. Through these two very different test programmes, the response and behaviour characteristics of a true nuclear facility were obtained.

Tests of fuel behaviour and damage have already been performed in various facilities around the world. Facilities such as PBF (US), Halden (Norway) and Phebus (France), are designed for testing one to four fuel rods in a test loop within a reactor. As such, these facilities do not test typical size fuel assemblies. LOFT is a complete, albeit small, PWR. Fuel tests performed at LOFT permit the damaging of a typical fuel assembly cross section and tracking the behaviour of the damage debris as it moves through the reactor coolant system.

The OECD LOFT Project fission product tests, designated LP-FP-1 and 2, were highly successful. The first test, LP-FP-1, was designed to take the fuel to the point of small ruptures in a limited number of fuel rods. The result of this mild test was the release of the radioactive gases contained in less than a dozen fuel rods. Retention of the material released by the reactor's cooling water and deposition of the material on the system piping and metal structures were much larger than predicted. The result is a much smaller amount of radioactive material that could potentially be released to the environment. At this stage, cleanup and preparation for more testing were not a problem. The second test, LP-FP-2, was considerably different, as it involved a major fuel damage. The criteria that would result in the damage levels desired demanded fuel temperatures in excess of 2 100 K (3 300 °F) for a period of more than three minutes. Failure to confine these high temperatures and protect the peripheral fuel assemblies would make it impossible to remove the central assembly for post-test examination, so it was necessary to design a special fuel assembly for the centre of the reactor.

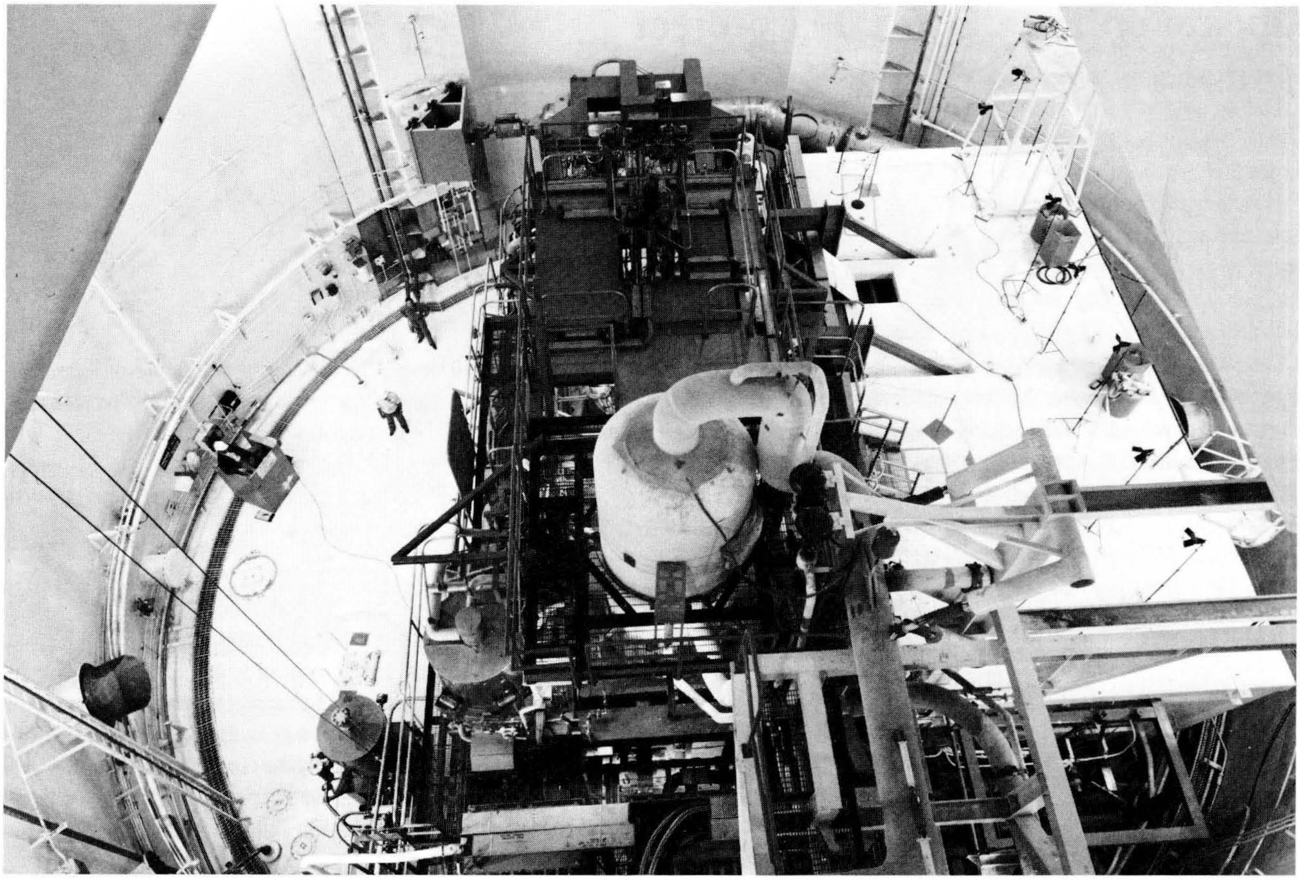
This last test released large quantities of radioactive material from the fuel rod and the fuel itself into the reactor coolant. Special systems sampled the movement of these materials throughout the reactor coolant.

Successful results

The first six thermal-hydraulic tests have provided information on the response of a PWR to various accidents and transients. The results provide data for assessing the capabilities of the complex computer codes used to analyse full-size commercial power plants as well as providing reactor operators with information to help identify certain transients and the phenomena that occur during the accident.

R.R. Landry is a member of the Agency's Nuclear Safety Division.

* Countries participating in the LOFT Project include: Austria, Finland, Federal Republic of Germany, Italy, Japan, Spain, Sweden, Switzerland, United Kingdom and United States. For more details on the OECD LOFT Project see the article in NEA Newsletter No. 3, December 1984.



Loss of Fluid Test Facility (LOFT): Looking down on the test assembly inside the containment vessel test chamber

The two fission product tests will provide data for both small ruptures and major damage on the release and transport of the radioactive materials contained in the fuel rods. They will also show the way in which a full cross section fuel assembly is damaged, a result that cannot be obtained from tests involving one to four rods.

The budget for the OECD LOFT Project exceeds US$ 80 million. As large as this may be, it is a small amount for a test programme as ambitious as LOFT. In fact, with the co-operation of the participating members the contractor was successful in staying within the budget. A large number of technical staff were provided by the Member countries and the post-test analyses are being performed and reported on by the Member countries. Moreover, staff members working within the contractor organisation gain first-hand, immediate knowledge of test results and they gain technically from the direct involvement in the planning and analysis of the tests. Post-test analyses and reports by the Member countries also provide experience from drawing together analyses from various members and explaining differences. The result is a deeper understanding of the tests, the meaning of the results and the analysis codes.

In view of the amount of information produced by the two final OECD LOFT Project tests, the LOFT Board of Management has asked the Nuclear Energy Agency to invite Member countries who have not yet acceeded to the Project to join in the analysis of the results of these tests ■

The Stripa Symposium: New findings and future directions

S.G. Carlyle

The primary option for the safe disposal of heat-generating highly radioactive nuclear waste is emplacement deep underground in suitable geologic formations. Research into the feasibility and safety of this option has been carried out by most OECD Nuclear Energy Agency Member countries for a number of years and it is an important part of the work at the NEA. This research has shown that, in principle, such disposal is both feasible and safe. To confirm this conclusion in practice it is necessary to conduct further research under realistic conditions. These *in situ* experiments within potential host rocks provide a means of: 1. developing tools for obtaining the detailed information required for site characterisation, 2. investigating phenomena which could influence the isolation capability of repository systems, 3. gaining experience in implementing and optimising the system's engineered features.

Detailed site-specific investigations in granite began in 1977 under the Swedish-American Co-operative Programme at the Stripa mine in Sweden and in 1980 evolved into Phase I of the NEA International Stripa Project. Operated by the Swedish Nuclear Fuel and Waste Management Company (SKB), the project is directed by a Joint Technical Committee representing the nine participating NEA Member countries*. During both Phase I and Phase II, which will end in 1986, work has been carried out in four main areas:

- Hydrogeological investigations of granite and tracer migration within single and multiple fracture systems;
- The detection and characterisation of fracture zones in granite;
- Hydrogeochemical investigations of granitic ground waters as represented by those at the Stripa mine; and
- The behaviour of bentonite clay as a backfill and seal material under field conditions.

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* Canada, Finland, France, Japan, Spain, Sweden, Switzerland, United Kingdom, United States.

Results to date

The results and conclusions from Phase I and preliminary results from Phase II were presented at the Stripa Symposium held in Stockholm in June, 1985. It is clear that the project has succeeded in developing new tools and techniques for assessing potential disposal sites in granite and that new insights into the performance of engineered barriers in the repository design have been gained. The ground-water flow studies at Stripa support the conclusion that there is no single technique for describing the hydrologic system around a repository adequately for detailed performance assessment. A combination of hydrogeologic, geophysical and geochemical techniques must be used.

Natural and artificial tracer studies at Stripa confirm that a small percentage of the fractures in granite transmit flowing ground water, as schematically shown in Figure 1. This has important implications for predicting potential pathways for radionuclide migration.* Fracture system characterisation, which by necessity

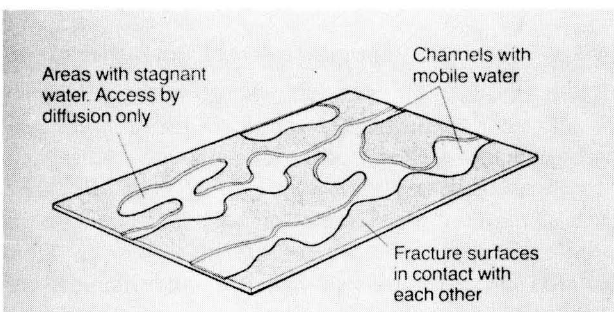

Fig. 1. Diagramatic Concept of Water Flow in a Fracture

* See the article in this issue by A.B. Muller on the NEA's involvement in the setting-up of international data bases on geochemical phenomena of importance in predicting the effects of migrating radionuclides through modelling.

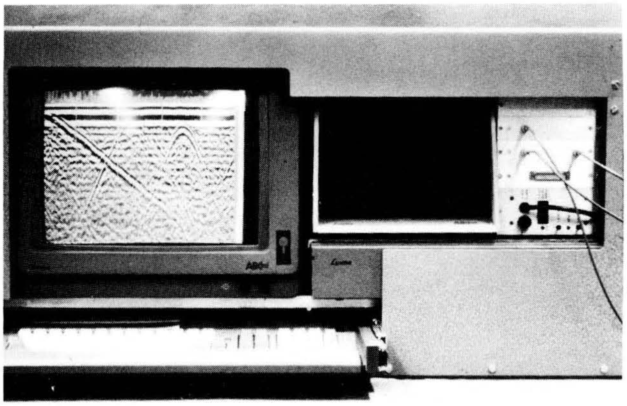

Fig. 2. Seismic Investigation Analysis using Tomographic Techniques

must use indirect means when dealing with large volumes of rock, is therefore significant. Crosshole tomography (three dimensional fracture mapping between boreholes) using seismic, hydraulic and radar techniques has been found to successfully locate more transmissive fracture zones. Figure 2 shows a typical seismic

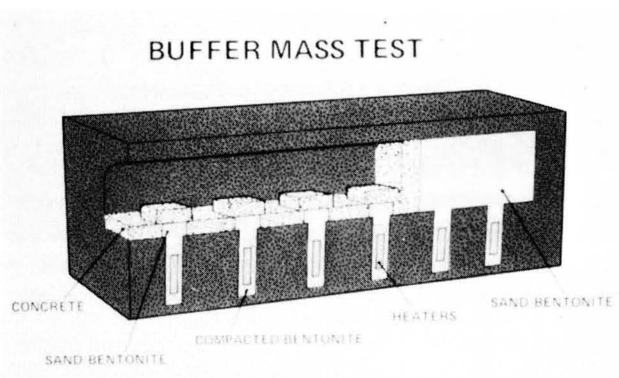

Fig. 3. Schematic Representation of the Buffer Mass Test Facilities

section being analysed using newly-developed tomographic methods.

Engineering studies at Stripa have concentrated on the validation of predicted bentonite behaviour as backfill and seal material in the Buffer Mass Test, shown in Figure 3. Results confirm that the primary physical processes occurring within the bentonite are understood and can be predicted for various repository designs and conditions. The most important process influencing the bentonite is its uptake of water from water-bearing fissures in the surrounding rock mass, since this governs the build-up of clay swelling pressure and influences the temperature dissipation from the waste.

Current work and future directions

Work currently underway in Phase II is designed to enhance the preliminary work performed under Phase I. Site characterisation techniques are being refined and their results correlated, tracer migration is being studied in a dynamic three-dimensional experiment and borehole and shaft sealing methods using bentonite are being evaluated. The results from these activities were presented at the Symposium, and have led to the formulation of plans for Phase III. This phase will concentrate on the application of techniques developed during the earlier phases in a large-scale investigation of an undisturbed volume of granite. First, a general characterisation will be carried out and predictive models will be developed. The resulting ground-water flow and tracer migration predictions will then be compared with field measurements in order to validate the models.

With this programme of work, Phase III of the Stripa Project will continue to provide fundamental information about granitic host rocks, train personnel in several NEA Member countries in new site characterisation and respository engineering methods, and contribute to increased public confidence in the deep geologic disposal of nuclear wastes ■

Geochemical data bases at the NEA

A.B. Muller

Deep geologic disposal is considered by most NEA Member countries the preferred option for the long term isolation of high level nuclear waste, either spent fuel or reprocessing wastes. These disposal systems can be considered to consist of a series of barriers, such as waste form and packaging, engineered repository and the geologic environment, which prevent the migration of radionuclides into the biosphere. Of these barriers, systems performance assessment modelling has shown that the geologic barrier generally contributes most to the long-term isolation capability of the disposal system.

The mobilisation and transportation of radionuclides by water is an essential element in any credible scenario for release of radioactivity from a repository. In some host geologic formations, such as massive salts or clays, water is not expected to be available for these processes. In others, water could eventually breach the waste package and dissolve radioelements. In such scenarios, chemical processes in the geologic barrier will, to a variable extent, retard (until some fraction has decayed) or actually stop the migration of radioelements in flowing ground water. Understanding these processes and being able to predict their effect through modelling are therefore essential for assessing the performance of the geologic barrier.

Distribution coefficient data

Geochemical modelling of radionuclide transport employs a number of approaches and methods, among the most important of which is the use of distribution coefficients (Kds) to model retardation. A distribution coefficient is an empirical parameter defined as the ratio of the amount of a radioelement in solution to the amount not in solution, in a given amount of geologic material (*i.e.*, concentration of a nuclide in solution divided by the amount lost into/onto the associated volume of rock). Removal from solution is generally taken to be by sorption processes, like direct adsorption or ion exchange, although due to the empirical nature of the parameter, other processes could be involved. Solutes would move through the geosphere at the speed of the solving ground water, which itself is extremely slow at sites selected for disposal, if no interaction would occur with the rock matrix. A more realistic velocity for dissolved radioelements in ground water, which will account for retardation, can be estimated by reducing the velocity of ground water calculated by a hydrologic model, by a term proportional to the distribution coefficient.

Distribution coefficients are measured in either batch or flow-through laboratory experiments designed to simulate, as closely as possible, the conditions expected in the geosphere. This is important, since distribution coefficients vary as a poorly-known function of these conditions, particularly for elements of multiple oxidation states, such as the actinides. Distribution coefficients seem most sensitive to solution acidity (pH), oxidation-reduction potential (Eh), major solute composition, the amount of radioelement present and sorption characteristics of the mineral phases. Detailed characterisation of these parameters during experiments is therefore essential to generating useful data.

In 1980, the NEA Radioactive Waste Management Committee recognised the usefulness of having a centralised international data bank of distribution coefficients for a variety of radioelements, geological materials and physico-chemical conditions. For this purpose, the NEA established the International Sorption Information Retrieval System (ISIRS) project in 1981. At the start of the project in June, 1981 the participating countries included Canada, Finland, Federal Republic of Germany, Italy, Japan, Netherlands, Sweden, Switzerland, United Kingdom and United States. During the first two-year period of the project a specialised data base management system for handling Kd data and the associated experimental parameters was developed at Battelle Pacific Northwest Laboratories (PNL) in the US. During the second two-year period, the system and its prototype data base were transferred to the NEA Data Bank in Saclay, France and it was evaluated and optimised by NEA and PNL staff. This has led to a third project period during which the system will be in routine operation.

The prototype data base with which the operational phase will begin contains data from about 2 500 Kd determinations for 18 elements on 8 general classes of geologic material under a variety of physico-chemical conditions. As the size and breadth of the data base increases during the operational period, ISIRS will be increasingly valuable for:

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- Providing distribution coefficient data for both general and preliminary site-specific models of potential radionuclide migration from disposal facilities;
- Providing experimental scientists with a readily accessible source of information about previous experimental parameters and results;
- Providing a collection of data from various sources that can be used to develop and validate geochemical models of sorption behaviour.

Chemical thermodynamic data

Another powerful tool increasingly used in the geochemical modelling of the far-field geologic barrier is chemical thermodynamics. Chemical thermodynamics permits the quantification of mass transfers in chemical reactions occurring in ground water and in water/rock interactions. The ultimate product of these models are predictions of the concentration of radioelements under various conditions. These models have also begun to be applied in the repository near field (where temperatures are higher) to waste glass and spent fuel dissolution and canister material degradation. Chemical thermodynamic models can be used to calculate:

- Aqueous speciation (the distribution of an element into the various ionic and unchanged species in which it can exist in solution);
- The mass of an element gained in or lost from solution, due to dissolution and precipitation processes;
- Mass transfer of solutes to/from solution due to sorption (ion exchange), solid solution formation, isotopic exchange and other retardation mechanisms.

The first two of these processes have been classically treated in chemical thermodynamic modelling while the third is currently under development.

Two types of data are needed to perform such calculations: data describing the specific solution and physico-chemical conditions for which the calculation is made and data describing the chemical reactions which may occur. The first type of data are calculation-specific while the latter are universal, in that the same data are used to describe the reaction in any calculation. These calculations are usually performed by speciation computer codes, such as WATEQ, or reaction-path tracking codes, such as PHREEQE. The NEA is now holding its second short course on the application of PHREEQE to nuclear waste disposal. Both of the codes were developed by the US Geological Survey.

In order to provide a broader variety of geochemical data to users in the Member countries, the NEA Radioactive Waste Management Committee established in 1983 the NEA Thermochemical Data Base and Critical Review as a complement to ISIRS. This data base contains fundamental chemical thermodynamic data from which the universal data base for speciation and reaction-path tracking codes can be established. The objective of this activity is to compile, critically review and publish recommended values of these fundamental constants for ten elements important to high-level waste disposal and other nuclear technologies. Uranium, plutonium, neptunium, americium, cesium, strontium, radium, technetium, iodine and lead are the elements under review. The results of the uranium review, which is the first volume to be published, will appear in early 1986. The recommended data sets will also be made available in directly computer-readable form.

By having established complementary international co-operative data bases for distribution coefficients (ISIRS) and chemical thermodynamic data (TDB), the NEA has addressed the primary geochemical modelling data needs of waste disposal systems performance assessment ■

New Publications from the NEA

The Economics of the Nuclear Fuel Cycle

ISBN 92-64-12714-3
Price: £ 12.50 US$ 25 FF 125 DM 56

Costs of separate stages and the total costs of the nuclear fuel cycle are reviewed for a once-through and reprocessing cycle of a PWR to be commissioned in 1995. Sensitivity studies are provided and some other fuel cycles are also examined.

The Management of High-Level Radioactive Waste
A Survey of Demonstration Activities, April 1985

Free on request

In order to assess the feasibility of high-level radioactive waste management technology, it is necessary to determine the soundness of management concepts and the adequacy of the technical experience gathered at the industrial level. This report summarises the achievements and on-going projects in OECD countries which contribute to the direct and indirect demonstration of the management and disposal of high-level waste.

Radioactive Waste Disposal: In situ
Experiments in Granite
Proceedings of an NEA
Symposium on the International Stripa Project,
Stockholm, June 1985

ISBN 92-64-02728-9
Price: £16.80 US$34.00 FF168 DM75

The purpose of the International Stripa Project is to develop techniques to investigate potential sites for the disposal of high-level radioactive waste and to examine particular engineering and environmental phenomena associated with the long term performance of a high-level waste repository. These proceedings present the results to date from the Project and from similar experimental facilities in the OECD area.

Review of the Continued Suitability of the Dumping Site for Radioactive Waste in the North-East Atlantic, 1985

Free on request

The continued suitability of the dumping site for packaged radioactive waste in the North-East Atlantic region was reviewed by an international group of experts in 1985. This report describes the basis for the assessment and the group of experts' conclusions and recommendations concerning the sea dumping operations.

Storage with Surveillance versus Immediate
Decommissioning for Nuclear Reactors
Proceedings of an NEA Workshop,
Paris, October 1984

ISBN 92-64-02724-6
Price: £ 14 US$ 28 FF 140 DM 62

A major question in selecting options for the decommissioning of nuclear facilities is whether it is more appropriate to dismantle a plant immediately or to delay this dismantling for decades or centuries. These Proceedings review the international debate on this matter, and report the consensus of a group of experts on the principal criteria to be considered in the selection of decommissioning options.

Metrology and Monitoring of Radon, Thoron and their Daughter Products

ISBN 92-64-12767-4
Price: £ 14 US$ 28 FF 140 DM 62

An important aspect of the control of radiation hazards from exposure to radon, thoron and their daughter products in mines and in dwellings is represented by accurate and reliable measurement of these radionuclides. This report reviews the principles and techniques of metrology and monitoring, and gives guidance on the selection and use of measurement methods and equipment for the various conditions of exposure of workers and members of the public.

A Guide for Controlling Consumer Products Containing Radioactive Substances

Free on request

Consumer products containing radioactive substances will cause some exposure to the public through use, misuse, accident or disposal. The purpose of this guide is to assist national authorities in defining the policy and basic radiation protection principles to be followed to ensure that radiation doses to the public from justifiable products are limited.

Continuous Surveillance of Reactor Coolant Circuit Integrity Proceedings of an NEA Specialist Meeting, London, August 1985

Continuous surveillance is important to assuring the integrity of a reactor coolant circuit. It can give advance warning of structural degradation and indicate where off-line inspection should be focussed. These proceedings describe the state of development of several techniques which may be used. They involve measuring structural vibration, core neutron noise, acoustic emission from cracks, coolant leakage, or operating parameters such as coolant temperature and pressure ■

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