NEA NEWSLETTER

SEPTEMBER 1984

SPECIAL ISSUE - NUCLEAR SAFETY

OECD NUCLEAR ENERGY AGENCY / 38 BD SUCHET / 75016 PARIS / FRANCE

OECD NUCLEAR ENERGY AGENCY

The OECD NUCLEAR ENERGY AGENCY (NEA) was established in 1972, taking the place of the European Nuclear Energy Agency (ENEA) which had been set up in 1958.

The 19 European Member Countries of the OECD have been joined in the NEA by Australia, Canada, Japan and the United States. The Commission of the European Communities (CEC) and the International Atomic Energy Agency (IAEA) both take part in the Agency's work.

The NEA works to promote co-operation between Member governments in the safety and regulatory aspects of nuclear power and in the development of nuclear energy as a contributor to economic progress.

This is achieved by:

- *reviewing technical and economic aspects of the nuclear fuel cycle;*
- *encouraging the harmonization of governments' regulatory policies and practices;*
- *assessing demand and supply and forecasting the potential contribution of nuclear power to energy demand;*
- *exchanging scientific and technical information; and*
- *co-ordinating and supporting research and development programmes, notably through the setting up of joint projects.*

Foreword

This special issue of the Nuclear Energy Agency's Newsletters has been prepared on the occasion of the ENS/ANS international meeting on Thermal Nuclear Reactor Safety, to illustrate by way of specific examples, the range of co-operative activities undertaken by NEA in the nuclear safety field.

It contains a selection of short contributions prepared either specially for this meeting, or recently published in recent regular issues of the NEA Newsletter.

Contents

Regulatory Trends in OECD Member Countries

by W. Dircks (USNRC)
and G. Naschi (ENEA)

on behalf of the OECD-NEA
Committee on the Safety of Nuclear Installations

ABSTRACT

At the beginnings of commercial nuclear power, national approaches to safety regulation tended to follow the strong U.S. lead. With the subsequent development and spread of nuclear technology, countries have since established regulatory practices specific to their own situation and no country now dominates the development of regulatory positions.

Even so, several recent common developments can be discerned, including: a natural transition in regulatory effort away from licensing of new plants to ensuring the safety of operating plants; greater emphasis on learning from operational experience; moves towards standardising plant designs and refining general safety criteria; redirection of safety research and development of new regulatory approaches to the severe accident question following the TMI accident; more attention to the human element in plant safety; use of the knowledge coming from large research programmes to develop new regulatory requirements; increased application of probabilistic assessment techniques to a broad range of regulatory problems.

The introduction of probabilistic approaches into safety analyses, and the program of international information exchange and joint assessment conducted under CSNI are both contributing to a better understanding of the common basis of nuclear safety regulation.

The future is likely to see more attention being paid to the problems associated with plant ageing, and a further convergence of approaches as safety research advances, operational experience accumulates and assessment techniques are further refined.

INTRODUCTION

Commercial nuclear power based on the light water reactor (LWR) was first developed in the United States. Accordingly, it was natural that for quite some time the United States had an unmatched influence on the development of the related regulatory safety philosophy. Other countries adopting the LWR often tended to follow the U.S. lead, by either modelling their own regulatory processes directly on the U.S. approach, or by at least attempting to develop schemes compatible with U.S. regulatory practice. However, these early regulatory requirements were developed when reactor technology was rapidly evolving, and when both reactors and the potential consequences of accidents in them were relatively small. Furthermore the first regulations had to be developed on the basis of a limited understanding of reactor behaviour in abnormal situations. Safety-related technical information was sparse since safety research was at an embryonic stage and little operating experience was available. What experience there was related to the first prototype reactors, and needed to be extrapolated greatly in order to predict the behaviour of the first generation of commercial plants which were five to ten times larger. As a result a pragmatic approach had to be adopted in devising early regulations. The need for caution made it necessary to incorporate conservative assumptions due to the gaps in knowledge. (The resulting conservatisms in reactor design were vividly demonstrated by the TMI accident, where weak points in certain areas were compensated for by the intentional over-design in others.) The size of industrial power reactors has since doubled again, and increasing technical understanding from research and operating experience has led to new and deeper technical insights into safety questions.

In recent years, rising expectations of individuals in industrial countries have led to continuing demands that large-scale industrial activities be conducted with <u>no</u> risk to the public. Nuclear power has borne the brunt of this mood, in large part because of the intangible nature of radioactivity and because of the military uses by which nuclear fission first came to public attention. This has contributed to increasing pressure for more stringent safety requirements, criteria and standards.

Several countries have now developed an independent capability in nuclear technology. Reactor vendors in different countries have produced various commercial LWR designs, which have been adopted by many utilities. Along with this spreading of nuclear technology, recent public debates subsequent to the publication of the U.S. Reactor Safety Study (WASH-1400) and the TMI accident have obliged each country to establish national regulations as a function of their individual situations.

All of these developments have combined to produce the current situation where no country clearly dominates the development of regulatory positions. When countries need to formulate new safety policies or practices, they are faced with a multitude of technical arguments and options as well as differing practices in other countries.

RECENT COMMON TRENDS IN LWR REGULATION

Development of Advanced LWR Designs and Refinements in General Safety Criteria

Notwithstanding this situation where diverse regulatory approaches exist, one can identify several recent trends common to most countries. From the very beginnings of commercial nuclear power, it was clear that the rate that new capacity came on line would eventually slow down as the industry matured. In consequence, regulatory authorities would have to shift their efforts steadily from the task of licensing new plants to ensuring that plants on-line were operated safely. The TMI accident has underlined a need now for greater regulatory surveillance of operating plants. Ever-increasing efforts are being made to feed lessons from the rapidly accumulating operating experience back to operational practice and regulatory activities. In several countries regulatory authorities supervise a systematic programme of periodic re-evaluations of plants throughout their operating life.

There is also a move towards evolving standardised advanced reactor designs, which should be both cheaper to build and more straightforward to regulate. This approach is now being followed up in several countries with major nuclear programmes, including France (the N4 1400 MWe PWR), the Federal Republic of Germany (the KONVOI scheme), Italy (PUN) and the United States (GESSAR 2, CESSAR System 80, RESAR SP/90 designs). Parallel to these schemes are regulatory efforts to refine general safety criteria, taking full advantage of the lessons learned from operating experience, the insights gained from safety research and the new analytical tools provided by probabilistic approaches to safety assessment. The new criteria include, for instance, revised requirements on the design, materials and inspection of primary circuit components, pressure vessels and steam generators, which are all aimed at reducing failure probabilities.

Long Term Regulatory Consequences of the TMI Accident

During the TMI accident the reactor core experienced transient conditions much more severe than foreseen in the design basis calculations. Even so there were no demonstrable offsite health consequences. The U.S. regulations in force when TMI was licensed were based on defining a "maximum credible accident" and demonstrating that its consequences could be contained. Any worse situation that one might conceive was considered to be so unlikely that it was classed as "incredible". This in turn meant that there was no need to take precautions against such events or their consequences. This whole approach had first been challenged publically by Reg. Farmer in 1967. Since no technology was free of risk, he pointed out, the mere fact of using nuclear reactors implied acceptance of some degree of risk. There was no logical way of differentiating between "credible" and "incredible" accidents. The 1975 U.S. Reactor Safety Study took this argument further and made the first quantitative estimates of the probability and consequences of the complete spectrum of conceivable LWR accidents. The lengthy debate following the 1979 TMI accident on whether or not the design basis had been passed led to reconsideration of the regulatory philosophy to adopt regarding "severe" accidents, i.e. those involving loss of key systems and eventual damage to the core. Although the TMI accident had the effect of concentrating the minds of the assessors on this issue, the result was limited to some changes in emphasis. The consensus now seems to be that resources should be devoted more to preventing severe accidents than to mitigating their effects.

The accident at TMI also showed that inadequate attention had previously been given to the role of the human element in avoiding and countering accidents. It pointed up a need to clarify management responsibilities, especially in relation to those of regulatory authorities, for assuring safety during both routine operation and emergencies. Also identified was a need to ensure that technical advice was available during an emergency, either by upgrading certain shift personnel or by engaging experts who were not directly involved in operating the plant. Regulatory authorities in different countries have since reviewed the role of management, the availability of technical expertise, and emergency planning for reactor accidents. Concentrated efforts are being made to see that provisions are improved where necessary.

The accident also brought to the fore the question of how reasonable it was to expect operators to act effectively following a serious perturbation to reactor operation, considering the stress imposed on them and the often inadequate and even contradictory information available to them. Many countries have since devoted substantial work on various aspects of the human factors question: redesign of control rooms in order to support operators in responding to an accident, design of emergency procedures in terms of the plant state to supplement those for each specific event sequence, provision of control room simulators for training staff to handle abnormal situations, and identifying extra equipment and making provision in advance to permit unusual system configurations that could enable operators to keep control of a situation involving core degradation.

Introduction of Probabilistic Techniques in Safety Analysis

In most technologies, safety requirements have been defined through an empirical trial-and-error method, often in the wake of accidents which have resulted in major loss of life. This approach was never considered acceptable in the nuclear industry. As noted above, extreme prudence in setting the first regulations led to substantial conservatisms in early plant designs, in order to compensate for the limited knowledge of reactor behaviour in unusual operating conditions. Regulatory organisations have recently devoted great efforts to develop and evaluate probabilistic methods of analysis suitable for making safety assessments. These new tools provide an additional means for assessing the level of safety in various parts of each plant and in the plant as a whole. It is becoming possible to demonstrate that balanced safety coverage is being maintained over the entire plant. In consequence, certain current safety requirements may turn out to be superfluous, while other measures may warrant reinforcement to remedy weak points. Even with the uncertainty inherent to global risk assessments, the risk-equalisation approach can provide useful indications of the safety value of various changes to reactor design or in its operation, and on how to set priorities on regulatory issues requiring attention or additional research.

The most comprehensive potential regulatory use of probabilistic methods is in the process of granting construction and operating licenses. Although the use of PRA is growing, it seems likely for the present that in most countries such studies will not be a required component of an application for a plant license. (In some countries a preliminary probabilistic safety assessment _is_ required at the early design stage.) Some countries have gone so far as to develop "safety goals" in terms of accident probability and consequences. Safety goal efforts can at best serve as sources of guidance to decision-makers for their evaluation of accidents beyond the design basis and the relative risk of nuclear and other sources of power.

Following WASH-1400 and TMI there has been a more systematic use of probabilistic techniques like fault trees and event trees, not only to evaluate and improve safety levels within a given plant, but also to compare the levels attained by different plants in the country, and even by plants abroad. Event tree analyses are making it possible to balance deterministically-based decisions with probabilistic insights. For instance, system reliability analyses are now being used to develop optimum programmes of quality assurance and preventative maintenance schedules.

The increasing use of probabilistic techniques is also helping to improve the common technical basis for regulation among OECD Member countries. For example, by the very nature of event trees they cannot incorporate the concept of inviolable barriers (nor of "incredible" sequences of events). As a result, it is generally agreed that the severe accident issue will not be resolved without the assistance of probabilistic techniques of analysis.

Contributions of Safety Research

Several large scale nuclear safety research programmes were begun in the 1960's and 1970's. These are now producing the answers to many of the long-standing technical questions about the safety of modern nuclear plants. The results of these studies and the introduction of the probabilistic approach to assessing safety are allowing regulatory authorities to fix their policies and requirements on the basis of measured physical phenomena and best-estimate calculations, rather than conservative bounding arguments. This should contribute towards a refinment of licensing approaches into a coherent streamlined whole from a series of positions taken on individual issues.

On the severe accident question, for example, as noted above, many experts agree that emphasis should be placed on prevention rather than mitigation. Recent research indicates further that radioative releases from severe accidents may be lower than predicted in the predominantly conservative assessments used in previous risk studies. Preliminary indications from this work indicate that this could have a substantial influence on the understanding of the consequences of severe accidents, hence emergency planning for them. Some countries are now applying the probabilistic approach to investigate the value of deliberate controlled venting of the containment atmosphere in certain accidents. By affecting the timing of the radioactive release, it is suggested, it should be possible to reduce both its size and the eventual consequences. However containments differ widely, and their detailed behaviour plays a key role in an accident sequence. Further work will be needed before the value of venting can be firmly established.

In parallel, following extensive experimental and theoretical studies there is less concern today that large steam explosions after a core melt present a realistic threat to containment integrity. A number of countries are continuing to carry out research into the behaviour of hydrogen generated from the zircaloy-water reaction, corium reactions and the behaviour of fission products inside the containment.

Evolution and Influence of CSNI Activities

The international co-operation on safety matters organised through the NEA Committee on the Safety of Nuclear Installations (CSNI) constitutes another means for limiting the current tendency to regulatory divergence. The evolution of CSNI's programme over the years has also reflected the trends in the preoccupations of national regulatory authorities. When the Committee was established in the early 1970's, its activities were concentrated on identifying safety research needed to improve understanding of reactor behaviour in accident conditions, the performance of safety systems and interactions between plant systems. CSNI has since been steadily shifting its efforts towards the comparison and joint assessment of research results and operating experience. This helps ensure that regulators in all countries have a common data base, as wide as possible, available to them. This fosters the development of technically coherent safety requirements in all countries (even if detailed designs will always vary somewhat for a variety of reasons). Following are several illustrative examples of recent developments in the CSNI programme.

In order to profit from the lessons of operating experience, all regulatory authorities have long-running programmes to collect and evaluate reports of safety-related occurrences. To maximise the benefits from these national schemes, CSNI established in 1981 an international Incident Reporting System (IRS) covering all OECD countries with operating power reactors (which represent about 80% of the world's nuclear capacity). Hundreds of incidents of particular safety interest have since been circulated through the IRS. As well as broadening the background information available to safety analysts for assessing incidents in their own country, the IRS gives regulatory authorities advance warning of potential safety problems and of the approaches being taken elsewhere to solve them.

Because of the strong influence that different countries' policies can have on one another, it is increasingly important for regulatory authorities to have regular opportunities to exchange views with their counterparts in other countries on important basic technical issues and regulatory practices. The CSNI sub-Committee on Licensing has held special meetings in recent years on several regulatory topics, including the backfitting of safety equipment, advance planning for nuclear emergencies, and the selection of sites for nuclear power plants.

The emergency planning case illustrates well how different practices can develop unrecognised. This area was the subject of a topical meeting of the sub-Committee in June 1981. The fundamental principles underlying emergency planning were reviewed, as were the practical measures that had been adopted or proposed in the different countries. Whilst all countries currently carry out emergency planning around nuclear plants, there were different views about its relationship with the licensing process. Considerable differences in actual practices were found, for example, in the scale and organisation of emergency planning, basic attitudes thereto, reference levels, the demands of the licensing process, the extent of public involvement, and so on. While some of these variations arose from countries' different traditions and administrative structures and were thus to be expected, others were technical in nature and primarily a result of current uncertainties in the source term.

In June 1983 the Nuclear Energy Agency convened an informal meeting of the Heads of the regulatory authorities from those countries having the broadest experience of industrial nuclear power development. The aim was to improve, at the highest level, mutual understanding of current trends in nuclear safety. The Heads of nuclear regulatory authorities acknowledged that in recent years there had been a growing tendency for OECD regulatory policies to diverge, and they undertook to work towards greater harmony, primarily through CSNI. As a practical step, they agreed to consult together in this framework as new major regulatory positions were being developed so that, first, some warning of impending changes could be given and, second, other countries could contribute to the process. These consultations have been implemented in the form of special meetings, two of which have been held so far.

The first was a special meeting convened early in 1984 to consider the regulatory basis for actions taken with regard to the problem of pipe cracking in boiling water reactors. After reviewing countries' experience with this phenomenon, the meeting went on to examine various technical aspects of detection and analysis of crack behaviour and how to mitigate the problem. It concluded with a general discussion of regulatory positions. The special meeting reached a number of conclusions: the phenomenon of inter-granular stress corrosion cracking (IGSCC) was well understood, low carbon material was preferable where stainless steels were used, and precise sizing of cracks during in-service inspection was a key safety factor in plants containing susceptible piping. Indeed the point was made strongly that the main reason for countries adopting conservative safety margins with regard to pipe failure was lack of confidence in methods of predicting crack growth and in the precision of ultrasonic sizing methods. The meeting went a long way towards clarifying the position regarding pipe cracking in BWRs, provided a great deal of information about what countries were doing and why, and reached consensus about the merits of the various interim solutions. As a result of the meeting there will be a fundamental technical compatability between the different approaches that the various national authorities come to adopt.

The second meeting, held in May 1984, took up the question of general safety criteria for advanced LWR designs. In some countries, most plants are of unique design, which means that regulatory authorities must review each of them in detail. The results are both high construction costs and a high level of regulatory effort required to license them. With the recent development of probabilistic techniques for safety analysis, a steady stream of results coming from safety research, and rapidly accumulating operating experience, regulatory bodies are now in a position to devise a new generation of general safety criteria that are more balanced and internally coherent. Such a development will contribute to more uniform plant designs, which will cost less to build and be easier to regulate. Consistent with this idea, the purpose of this meeting was to promote more coherent regulatory approaches among Member countries.

The meeting first reviewed the current programmes in France, the Federal Republic of Germany, Italy and the United States to develop standardised advanced PWR designs, along with related efforts in these and several other countries to revise existing general safety criteria. Several specific issues related to safety criteria were then singled out for further discussion. For example, it was noted that the United States was the only country to have formulated quantitative safety goals, and even there, these were being implemented on a limited trial basis. Most other countries were waiting for the results of the U.S. trial before proceeding much further in this area. Discussion of the degree to which operators are allowed to intervene during emergency situations revealed that the underlying philosophies in each country were not as dissimilar as the differences in the formal criteria would imply. Significant differences were found in the single-failure criteria in use in various countries. It was noted that whereas a single-failure criteria could be stated in rather general terms, its application was often quite complex. Even though probabilistic analyses of the current practices might show that there were no significant differences between them as regards risk, there was no clear picture of the philosophies on which the different practices were based. Concerning the question of anticipated transients without scram (ATWS), the meeting found that many countries were in the process of formulating or finalising new criteria. Approaches under consideration included efforts to reduce scram frequency and improve the reliability of existing scram systems, or a requirement for diverse actuation - or even complete - scram systems.

The meeting was very useful for identifying the fundamental philosophies of countries on a broad range of current interrelated questions. Future meetings will delve deeper into the reasons underlying the observed differences in several specific areas, and it will be valuable to take into account industry views in these discussions.

FUTURE OUTLOOK

The rules and regulations governing nuclear power programmes are necessarily complex. As illustrated above, the original U.S. lead in LWR regulation has given way to the current regulatory situation in which each country conducts largely independent activities reflecting its own particular situation. The natural tendency for these parallel efforts to diverge is being limited by the widespread introduction of probabilistic approaches to safety assessment - a trend given added impetus by WASH-1400 and TMI, and by the international collaboration organised through CSNI.

Nuclear regulation has come a long way from the initial attempts which had to incorporate substantial conservatisms to compensate for limited basic technical knowledge of the time. Regulation is becoming more coherent and balanced as a result of ever-increasing understanding from research and operating experience and the availability of more powerful assessment tools. Regulators are using the most modern analytical techniques and research results available in order to improve understanding of reactor behaviour and to make safety assessments as efficient and definitive as possible. Along with the development of standardised advanced LWR designs, work towards the development of general safety criteria should also encourage further convergence of different countries' approaches.

As stated earlier in the paper, there is a clear trend whereby regulatory authorities are applying the lessons from operating experience to improve regulatory processes. New questions will certainly arise in connection with plant ageing, and the regulation will need increased flexibility in order to devise plant-specific interim remedies to enable continued operation with adequate safety margins, quite apart from long-term solutions which may need several years to develop and implement.

It should always be kept in mind that by the very nature of the regulation function, the responsable authorities cannot take the lead in nuclear power development. Regulatory bodies can only strive to be responsive to advances (such as standardised plants), novelties (such as controlled venting), and proposed technological changes (such as improved instrumentation). It is up to industry to identify ways to improve plants and to see to it that they are operated in a safe manner. The nuclear industry must always remember that dedication and will on the part of all involved is indispensable to keeping nuclear power safe.

The Role of the Severe Accident in Nuclear Safety

by Klaus Stadie
Deputy Director, Safety and Regulation, NEA

The question of severe accidents poses a major challenge to the nuclear industry and efficient international cooperation is needed to maintain an international consensus in nuclear safety. This article describes some of the basic changes that have been taking place in safety philosophy and the contribution of the NEA in this field.

The application of any technology incurs certain risks. Early technology posed risks primarily to its workers and rarely, if ever, involved the public at large - with the exception of transport risks where the role of the operator and the customer (public) are inextricably linked. Only with the advent of modern technology, such as important hydro dams and large chemical complexes have people totally unconnected with the enterprise been threatened by accidents in these facilities.

As people have become more aware of this risk they have demanded more say in the continuing debate on what levels of risk are acceptable, a decision that has higherto been left to the wisdom of technical experts representing both industry and government, which normally regulates the use of technology.

Unfortunately, as we know, risk acceptance is not a strictly rational process and public attitudes to risk are affected by many powerful influences, including questions of utility and psychological factors. As far as the latter is concerned, we must also recognise differences between voluntary and involuntary risks.

If we take, for example, the case of the quite staggering figures of the road toll (a voluntary risk) in the industrial countries, (more than 120,000 killed every year in the OECD countries), we see that public perception of the dangers of motoring have had very little effect on car ownership. When we look at the reverse side of the coin, taking the example of nuclear power (an involuntary risk), we notice that public opinion in many Western countries is completely mesmerised by the ultimate potential for damage posed by a nuclear reactor, despite the evidence that a catastrophic accident is not in the least likely. This attitude showed up clearly in the aftermath to Three Mile Island (TMI) which - we must recall - did not kill a single person.

Defence in depth

This is not to play down the small, but very real, possibility that a nuclear power reactor could fail, with serious effects on people, even some distance away. But it may help to explain the difficulties the regulatory authorities are faced with when attempting to decide acceptable and rational levels of protection. In fact the risks inherent in exploiting nuclear power have been known from the outset and safety considerations have always played a major role in nuclear development, bringing it to the point today where it can be described as perhaps the safest industry. Not a single person of the general public has yet been killed as a result of an accident in a nuclear power reactor in the OECD area.

It is therefore doubtful whether it is wise to go a great deal further to provide costly devices designed to mitigate the consequence of extremely rare accidents. The economic penalty to the industry of following this course would be enormous compared with the very small reduction in risk accrued and it would be difficult - if not impossible - to verify the proper functioning of these devices during such an accident.

This argument has to be seen in the light of the fact that everything humanly possible has already been done in the nuclear safety field to cope with human fallibility through a defence in depth system, consisting of several layers of redundant and diverse safety devices. These different layers give assurance that we can limit the progression of any accident, the initiation of which in the first place is made at the very least unlikely by a vast quality assurance programme.

Safety approaches

So why raise the question of acceptable risk levels ? The answer can be seen in the way safety thought has developed since the 1950s. Up to now the fundamental safety approach for the most common reactor type in the OECD area - the light water reactor - developed in the United States where the type originated. This approach was - and is - based on the definition of a maximum credible accident and the demonstration that its consequences can be contained. Anything worse than this type of accident is considered so unlikely that it is classed "incredible", which in turn means that there is no need to take precautions against these events or their consequences. This concept, or a similar one, has been adopted in all OECD countries with water reactor power programmes.

In 1967 Mr. Farmer publicly challenged for the first time this black and white approach to nuclear safety. He pointed out that the mere fact of using nuclear reactors implied the acceptance of some finite degree of risk and, since no technology was entirely risk free, there was no logical way of differentiating between credible and incredible accidents. The 1975 Reactor Safety Study took this further and made the first quantitative estimates of the risks associated with nuclear power production in light water reactors. Later on many similar studies in other OECD countries have followed the WASH-1400 lead.

Finally there was the Three Mile Island accident which, depending on the way you look at it, was a credible or incredible event, according to the twenty year old definition of the design basis accident (DBA). TMI therefore gave additional impetus to the search for a new policy.

International cooperation through the NEA

The need for a common policy on accidents was perceived some time ago by the NEA Committee on the Safety of Nuclear Installations. This Committee directs a broad and comprehensive international programme in nuclear safety technology and licensing. In 1980 the Committee established a group of senior experts from a number of Member countries to study the potential response of existing water reactor safety systems to class 9 accidents (accidents beyond the DBA) and to examine the implications for current safety research and development.

It soon became clear that there was no common understanding of what constituted a class 9 accident - a term which in any case dates back to the time when the DBA was defined. Our experts finally decided to use the term "Severe Accident" instead, defining this as an event in which there is a failure of structures, materials, systems, etc., without which core cooling cannot be properly assured by normal means.

The Senior Group agreed unanimously on a number of questions. In general there was a consensus that the capability of PWRs to protect the public was far greater than the DBA approach implied. It was agreed that the first priority in safety should be accident prevention and then its progression thereafter to successive stages. At the same time the undoubted ability of plants to function safely beyond the DBA should be turned to account so as to maintain maximum control over events and thus keep the potential hazard to a minimum.

The most important area of concern for the experts was accident management, extending throughout the sequence of events making up a severe accident. Highest priority was placed on improving the ability of plant personnel to cope with severe accidents. It was felt that under this heading of accident management, a number of actions called for urgent attention: research aimed at producing best estimates of accident sequences, identification of key parameters in accident progression to enable appropriate instrumentation to be developed, the training of operators to diagnose severe accidents in terms of physical phenomena not scenarios, and more study of long-term accident management.

This work continues. We see as a priority task the need to develop a safety rationale which adequately takes into account accidents which have very low probability and at the same time potentially high consequences, without requiring absolute evidence that these events do little or no harm.

Much more work still needs to be done to arrive at a common policy. The OECD, and in particular its Committee on the Safety of Nuclear Installations, is pursuing this question vigorously, because we are clearly aware of the difficulties Member countries will encounter in their nuclear power programmes if we do not reach a consensus on how to treat severe accidents.

Assessing the Human Factor in Nuclear Safety

by Michael Stephens
Nuclear Safety Division, NEA

"The nature of the human factors question, the reasons why it is difficult to estimate its importance to nuclear safety, and some of the work being done to resolve it."

A tired workman misreads a poorly printed label and turns off the wrong control system. An operator misunderstands an instruction and decreases rather than increases a coolant flow. During a shift change a technician forgets to tell his successor how a certain circuit is set up and circuit breakers are not reset after a system check. These sorts of errors and omissions occur in any industrial plant, including nuclear reactors. How important are these mistakes for the safety of the plant ? How can we minimise their frequency and consequences ? How do we assess their impact on system reliability and availability ?

It is difficult to evaluate human performance even qualitatively because humans are not like machines; people may act in many different ways, at different times and their decisions are affected by many psychological factors. For instance, people may perform even well-defined tasks differently and with greater or less efficiency, depending on how familiar each is with the task, how tired each one is, what other tasks each has to perform, their respective understanding of the task and the specific objectives to be reached, the changing physical environment at work, their own social relationships, and so on.

Another complicating factor to add to this lengthy list is the large degree of interdependence between an individual's different actions. Once someone makes a mistake, he or she is quite likely to repeat it or even to compound it with further errors. It is also very difficult (if not impossible) to define an "optimum" way of doing a task. It is more useful to define an acceptable range of performance during a given period, clearly identifying the goals to be reached and the limitations and assumptions made in describing the human performance expected.

The first determined effort to measure the importance of the human element in nuclear safety was part of the 1975 US Reactor Risk Study, commonly known as WASH-1400. [Reactor Safety Study, USNRC, NUREG-75/014 (1975)] But this study had its shortcomings. The principal models to describe human acts available at that time assumed that there were "typical" errors that people tend to make, specifically: errors of commission (a needed action not done correctly), omission (an act not done at all), timing (acts not completed within the required time), sequencing (acts out of order) and extraneous (unnecessary acts which interfered with the normal sequence of events). The psychological reasons underlying such errors were not treated in any detail.

An NEA Contribution

The limitations of this approach, both in data and modelling, were soon pointed out, and in 1978 the NEA Committee on the Safety of Nuclear Installations (CSNI) launched a survey of the available sources of information on human reliability to study how assessment techniques could be improved.

The first area for review took in the many descriptions of incidents in nuclear plants being submitted to various national reporting schemes. These data and information had the advantage of being "real"; however, since the nuclear industry has a relatively short history the statistical base was inadequate. Besides, human actions aggravating incidents were commonly described in these reports very incompletely, often simply as "human error". Important details were usually missing, for example the time at which an error was made and when it was later discovered; other factors contributing to human errors were rarely reported (e.g. what the person involved was trying to accomplish, what necessary information was missing, what false assumptions he was led to make in trying to decide how to reach his goal).

Several other possible sources of human performance information were surveyed by the CSNI group. For instance, both military and aerospace organisations have been collecting human reliability data for a long time. However this information is highly specialised and not readily transferable to the personnel in a nuclear power station. Another potential source of information on operator performance was the record of training sessions carried out on control room simulators. Unfortunately the divergences between reality and the simulator situation made this kind of data suspect.

Even attempts to collect "human reliability data" lacked a way of putting mistakes into clear categories. Thus if one tried to collect information on specific incorrect human acts (e.g. misreading labels), there remained the problem of specifying the psychological context in which each error was made. Apparently simple statistics could be collected, such as "how often large valves are left in the wrong position in a typical plant". However, as noted above, there are many possible psychological reasons for making such errors, so that the statistics could not help much in finding ways to correct the causes.

A Description of Human Behaviour

What could be done to overcome these obstacles ? The key to making any progress lay in agreeing on a coherent, qualitative description of the psychological mechanisms behind human acts and the factors influencing them. Once this was established, then it would be possible not only to collect consistent human reliability data, but also to deduce ways of improving system performance and describe the importance to safety of the remaining errors that had to be expected and allowed for. A three-level model of human thought processes was adopted, defined in terms of trained skills, learned rules and creative thought. This distinction between modes of thinking is important because an incorrect act can have resulted from quite different types of mental error. For instance, the person involved may be clumsy, may forget that the situation is a special case, or may incorrectly interpret what is occurring.

The NEA Group drew up a set of data categories based on this way of describing thought processes gone astray and the associated casual factors (e.g. misleading or missing information, extreme physical conditions, emotional stress, etc.). The Group also developed a strategy to make the best use of available information sources, to collect human behaviour data in a more efficient manner than any of the existing schemes, and in a form directly usable by human factors analysts. This strategy involved asking plant personnel routinely to report background information on incidents, and having teams of specialists review plant records and interview personnel on selected important events to analyse them in more detail.

Improving the Human Factor in Nuclear Safety

To improve safety it is more profitable to optimise the environment in which a skilled person has to work than to attempt to "improve" his or her intrinsic capabilities in some fashion. Even a shallow understanding of human behaviour makes it possible to deduce some general principles aimed at helping humans perform well. Many of these principles are as obvious as they are commonly broken: maintain distinctive, consistent labelling of equipment, control panels and documents; design systems to give unambiguous responses to human intervention; and design systems to overcome failures due to human causes (or minimise their consequences).

The Three Mile Island Accident

It has often been asked whether safety analyses prior to TMI had included the events that occurred there. Similar sequences of equipment failure had in fact been considered in previous risk assessments. However the specific sequence of human actions involved had not, as one might have expected, been singled out.

However, focussing on the errors that were made at various stages of the accident (with the benefit of hindsight) one can see that certain conditions at the plant favoured the kinds of errors that were made. For example, several equipment deficiencies led to the staff receiving incorrect information at various times: repair tags concealed indicator lights on the control panel; one indicator showed a valve to be shut when in fact it was stuck open; one gauge was on a secondary control panel out of direct view. Other equipment deficiencies complicated the situation: two valves had wrongly been left shut after an earlier system check; a persistent minor leak disguised the occurrence of a later large leak. The operators' previous training led them astray as well: they stopped coolant circulation pumps after misinterpreting the significance of coolant levels. Subsequent assessment of the whole TMI train of events has led to significant improvements not only in equipment and training practices, but also in how the human element is now viewed and allowed for in plant design and operation.

Human factors studies are now advancing rapidly in many countries. Much greater attention is being paid to human needs in designing equipment people must operate, and in learning from experience to correct the errors of the past. Much effort is being devoted to developing mathematical techniques for quantifying the importance of human performance to the reliability of systems and the risk from accidents. Current NEA work is focussed on how to train operators to understand better what is happening during plant emergencies, and on the procedures and equipment that can be provided to them to cope with accidents where the reactor core has been damaged or vital systems impaired.

Copies of the reports prepared by an NEA Group, "A Guide to Writing Maintenance, Test and Calibration Procedures" (CSNI Report No.68) and "Assessing Human Reliability in Nuclear Power Plants" (CSNI Report No.75) can be obtained by writing to the Nuclear Safety Division, OECD Nuclear Energy Agency, 38 boulevard Suchet, F-75016 Paris, France.

The Reliability of Non-Destructive Test Methods: The PISC Programmes

by Peter Oliver
Nuclear Safety Division, NEA

"The problems of stress corrosion cracking in BWR pipes, or the cracks found under vessel cladding in French PWRs are recent manifestations of a long-standing general safety issue.

What is the outlook for non-destructive methods for detecting, locating and sizing flaws ?"

The origins of PISC

A number of OECD countries joined an informal international collaborative project in the late seventies, organised in the "Plate Inspection Steering Committee" (PISC) to assess the limits of accuracy of crack detection methods. The impetus for the PISC programme came from the growing international understanding of the propagation of cracks in steels, and its consequences for the integrity of a structure, and also from the developments in detection methods. The project was a straightforward international test of an ultrasonic test procedure recommended in the American ASME XI code and used in many countries.

Three thick (30 cm) steel plates containing welds were made available by the United States Pressure Vessel Research Committee (PVRC) long-term research programme and were shipped in turn to ten OECD countries, where inspection teams attempted to locate the artificial flaws which had been implanted in the welds, using the ASME procedure. The teams were encouraged also to inspect the test plates using any other advanced methods.

The inspection results, when compared with the real defect distribution revealed when the plates were cut up at the end of the programme, showed considerable dispersion between the different teams. They also showed that standard test procedures would find only about half the kinds of defects which under the rules of theoretical mechanics were thought to be undesirable in a typical PWR vessel. (Other methods gave better and more consistent results). This first PISC programme was in no way a scientific experiment, but it did draw attention to the need for proper training of inspectors to improve consistency of results, and to the urgency of bringing in newer techniques capable of locating all significant flaws.

Locating flaws in thick steel sections

The NEA therefore decided to run a second PISC project in order to examine more closely how reliably the best available methods could locate and size flaws in thick steel sections. Four new test blocks - two flat plates containing butt welds and two with set-in nozzles - were provided by four Member countries (Federal Republic of Germany, Italy, Japan and the United Kingdom) for inspection in an international round-robin test. The plates, weighing up to 16 tonnes, have various defects implanted into the weldments. They are being circulated to 14 countries in a three-year scheme, which is due to be completed in August 1984.

Inspection teams with no knowledge of the defect patterns are being encouraged to use any ultrasonic inspection techniques and procedures they choose as long as all the information is submitted to the "Referee Laboratory" along with the test results. The data are being computerised for comparison with the actual flaw pattern which will only be known when the plates are sectioned in a "destructive examination" at the end of the year. Data are coded to preserve the anonymity of the test teams, the identities of which are known only to the Referee Laboratory.

In parallel with the round-robin trials, a number of special research laboratory studies are being carried out in five Member countries to determine quantitatively the effects on defect detection and sizing of a number of factors, such as the presence of stainless steel cladding on the plates, the characteristics of the test equipment, and the precise nature of the defects themselves. These studies will end at the same time as the round-robin trials, and the two sets of results will be combined to give further insights into the question of reliability.

Great strides have been made in ultrasonic inspection methods over the past decade: automatic recording of probe movements and computerised data handling combine to make inspection much less dependent on the operator and thus consistency has been improved. The PISC-II project will point the way to the best procedures and establish for the first time how results are affected by factors such as cladding and equipment variables. The results will be brought to the attention of regulatory and licensing authorities as a contribution to the development of improved codes of practice.

Learning from Experience: The NEA's Incident Reporting System (IRS)

by Koichi Morimoto
Nuclear Safety Division, NEA

"For a long time many OECD countries have been looking for better ways of exchanging information on incidents in reactor operations in order to feed the lessons learned through experience back to the design and construction stages."

Amongst the Member countries of the NEA there has been a regular exchange of information on significant operating experience since 1965, though it was seen that a more systematic and efficient way of exchange was needed as the number of nuclear power plants in each country increased and their equipment became more standardized. The TMI accident in 1979 helped to push the demand for such a system. The NEA involvement began in 1980 when the NEA Committee on the Safety of Nuclear Installations (CSNI) introduced the Incident Reporting System (IRS) on a trial basis. The participating countries took a further step forward in 1983, by agreeing to work out guidelines and accept exchange procedures and reporting thresholds. In the past twelve months the System has been brought up to full operation.

The IRS has two main aims:

- Firstly it enables regulatory authorities and utilities to benefit from the lessons learned from significant incidents which happen in other countries. They are then able to decide whether the lessons should be applied to their own plants and how to take corrective actions to prevent similar incidents.

- Secondly, it can help identify areas of concern or generic safety issues where further improvements in system designs or operational practices and safety research are considered necessary.

Countries taking part in the IRS commit themselves to reporting incidents in accordance with common reporting procedures and criteria. An NEA country with no nuclear power plants can join the system as an observer and receive the circulated information, provided it agrees to enter into a reporting commitment if it eventually takes up the nuclear option.

All communications between the participating countries and the NEA Secretariat are handled by "IRS Co-ordinators" chosen by local authorities. Co-ordinators have two main tasks. They screen events according to safety significance and prepare reports. Every IRS report carries a cover sheet which briefly summarises the incident. The description of the events, possible causes, any lessons learned and actions taken are set down in detail. The "Reasons for Reporting" are also included, to indicate why the Co-ordinator regarded the incident as significant. Completed reports are forwarded to NEA in Paris.

Co-ordinators also circulate IRS reports, received from the NEA Secretariat, to appropriate organisations in their own country such as utilities and research institutes.

Reporting criteria is fairly comprehensive, special attention being given to: significant degradation of safety-related systems, release of, or exposure to, radioactive material, failures in design or construction, the effects of unusual external events, and also events which attract "significant" public interest.

Data retrieval

A computerised data retrieval system is now being developed in co-operation with the Ispra Establishment of the Joint Research Centre of the Commission of the European Communities to help IRS members and the NEA Secretariat to search for specific items through the incident reports and to provide information for assessment.

During 1983 a preliminary survey was made of the 235 incidents which had been stored in this data bank. This computer-aided analysis identified the qualitative trend of the incidents from a number of different aspects, such as the significance of human errors, the identification of failed systems, the classification of incidents according to their characteristics and the co-relation with plant ages. This survey showed that a number of important aspects useful for the safety analyses of incidents could be identified with the help of a good data retrieval system. Further improvements of the software and of the methods used for analysis are now being undertaken.

Besides this useful analysis being carried out at Ispra, NEA is also investigating the possibility of enabling participating countries to be given direct on-line access to the information stored in the data bank. To identify any technical difficulties and to evaluate the costs, the Commissariat à l'Energie Atomique (CEA) in France is setting up a trial on-line link to the computer system and, based on this experience, a proposal will be prepared for discussion by the experts in participating countries.

Once a procedure for on-line access is decided, each country will be able to establish a contact with JRC Ispra and search the data easily and quickly. As the number of incidents stored in the system increases, the data bank will become a much more effective tool and perhaps eventually an indispensable instrument for safety assessment.

Relationship with the IAEA system

The International Atomic Energy Agency (IAEA) has also been planning a similar incident reporting system for the IAEA group of countries and there has been some discussion on ways of exchanging information between the systems to avoid duplication. The importance of demonstrating reciprocal contributions and of assuring the confidentiality of reports has been agreed by all the participating NEA countries.

Besides this exercise the two organisations recently arranged a meeting for information exchange on operational experience in Paris attended by experts from 18 countries, including five non-NEA countries - Czechoslovakia, India, Republic of Korea, USSR, and Yugoslavia. Topics such as stress corrosion cracking, valve malfunction and mispositioning, independence in electrical and control systems, frequency of unscheduled shutdown, and the reliability of emergency diesel generators were highlighted as important aspects to be analysed in more detail. The participants welcomed the chance to exchange operating experience freely and directly, and to discuss the lessons learned from significant incidents on a world-wide scale.

For the next meeting in July 1984 emphasis will be put on the exchange of detailed information rather than the assessment or analysis of incidents. The selection of significant incidents and the detailed analysis of generic safety issues will be dealt with separately at the annual CSNI meeting.

Improvement in the rate of reporting

By March 1984 a total of 518 reports covering 456 incidents had been circulated through the NEA-IRS. The reporting rate has been considerably improved and is now averaging around 0.6 incidents/reactor year. The variance in contributions between countries has been brought down and IRS is now coming to be considered as one of the most successful international projects in the areas of nuclear safety. Further development of the system will help to enhance the safe operation of nuclear power plants in all the OECD countries.

Highlights of the NEA Programme on Nuclear Safety and Licensing

INTRODUCTION

NEA's programme has always included substantial work related to the safety of nuclear power. Originally focussed only on radiation protection aspects, this developed in the 1960s and early 1970s to include a large number of questions in the field of reactor safety technology. With more power reactors coming into service and the increasing emphasis on health and safety, OECD countries became more concerned with the problems of operating their nuclear power stations in a safe and reliable manner, and over the years the NEA has taken care to continually adapt its working structures to meet these needs.

The purpose of the NEA Committee on the Safety of Nuclear Installations (CSNI), which is an international committee made up of scientists and engineers who have responsibilities for nuclear safety research and nuclear licensing, is to foster international co-operation in nuclear safety amongst the OECD Member countries. This is done in a number of ways. Full use is made of the traditional methods of co-operation, such as information exchanges, establishment of working groups, and organisation of conferences. Some of these arrangements are of immediate benefit to Member countries, for example by enriching the data base available to national regulatory authorities and to the scientific community at large. Other questions may be taken up by the Committee itself with the aim of achieving an international consensus wherever possible. The traditional approach to co-operation is increasingly being reinforced by the creating of co-operative (international) research projects and the organisation of international standard problem exercises, for testing the performance of computer codes, test methods, etc. used in safety assessments; these exercises are now being conducted in most sectors of the nuclear safety programme.

The greater part of the CSNI co-operative programme is concerned with safety technology for water reactors. The Committee has set up five international "principal working groups" to manage its co-operative activities in the areas considerd to be of greatest importance: operating experience and human factors, reactor transients and primary circuit breaks, the integrity of pressure vessels and pipes, the phenomenology of radioactive releases in accidents and their environmental consequences and, finally, assessment of the associated risks. CSNI also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on power plant incidents.

With some 240 power reactors now in service throughout in the OECD area and another 130 under construction, it is neither practical, nor desirable, for research issues to be seen as separate from licensing research questions. Many technical problems dealt with in the Agency's nuclear safety programme are intimately bound up with regulatory requirements. Thus representatives of national regulatory authorities are increasingly involved in the CSNI programme, resulting in a perceptible move towards an integration of the research and licensing sides.

INFORMATION ON SAFETY RESEARCH

The Agency's information exchanges on nuclear safety research are based on the Nuclear Safety Research Index, published regularly by the NEA since 1971 and now issued every two years jointly with OECD's International Energy Agency (IEA). The 1982 Index listed standard details of some 900 current research projects and about 300 computer codes all related to nuclear safety. The Index is widely distributed to interested groups in Member countries, including government bodies, public and private research institutes and universities. A new Index will be published towards the end of 1984.

OPERATING EXPERIENCE AND HUMAN FACTORS

Operating Experience - Incident Reporting System

In 1980 the Agency set up the Incident Reporting System (IRS), to collect and disseminate information on operating experience in nuclear power plants. The system is now regarded as fully operational; about 200 incident reports per year are circulated through the IRS at the present time (it is to be noted that there were no incidents in nuclear power plants in OECD countries which demonstrably affected the health or safety of the general public). The IRS scheme functions under a Recommendation of the OECD Council, and covers all thirteen OECD countries that have nuclear power plants in service. During 1983 and 1984 the process of extending the coverage of the IRS took a step forward with an exchange of reports from non-OECD countries through the International Atomic Energy Agency (IAEA). From the different incidents reported nine current generic issues with major safety significance were selected for in-depth analysis. These included, for example, stress corrosion cracking, valve mispositioning, and the reliability of emergency diesel generators.

In the future, much greater effort will be devoted to assessing reported incidents for safety significance, ranking them in order of importance, and interpreting them in order to point out the lessons to be learned. To facilitate retrieval of the rapidly growing amount of information exchanged through the IRS, an IRS Data Bank has been set up at the Euratom Joint Research Centre of the Commission of the European Communities, Ispra (Italy).

Human Factors in Nuclear Safety

Analysis of incidents in nuclear power plants, not least the Three-Mile Island accident of 1979, clearly shows that the possibility of human error in operations must be assessed and allowed for if plant safety is to be maintained and enhanced. While Member countries are working to reduce the occurrence of operator errors, for example by improving operating procedures, an equally important aspect is the study of human error itself. A group of experts has completed an initial review of the various approaches to structuring emergency operating procedures in Member countries.

REACTOR TRANSIENTS AND PRIMARY CIRCUIT BREAKS

Any assessment of nuclear plant transients depends heavily on an understanding of the thermal and hydraulic phenomena taking place in the reactor. Clearly, a purely experimental approach to assessing plant response to such transients is not feasible, and in its place numerous and complex computer programs have been developed to calculate power plant response to different transients - and hence accident sequences - for different power plant designs. The results of these calculations provide the basis for decisions about the design of emergency core cooling systems (ECCS).

During development of these computer programs, their principles are related to experiments about individual phenomena (separate effects tests) and the behaviour of complete systems (integral tests), conducted in different facilities built to various scales. Comparison of predictions with separate effects tests ensures that individual phenomena have been properly modelled. Comparison with integral tests shows that system interrelations are correctly handled. It is also necessary to confirm that the computer programs can correctly extrapolate the phenomena observed in experiments to the full scale power plant system.

In order to compare computer program predictions with the results of related experiments, CSNI began a series of international standard problem (ISP) exercises in 1975. Most of the initial ISPs were related to loss-of-coolant experiments. So far, 17 standard problem exercises have been completed; four further exercises are proceeding, dealing with problems of fuel behaviour, loss-of-coolant accidents and containment response.

As far as ex-vessel thermal hydraulics are concerned, some fundamental uncertainties remain. A group of experts is identifying and quantifying from a risk perspective important physical phenomena that dominate core melt processes.

OECD LOFT PROJECT

The Loss-of-fluid test (LOFT) facility, Idaho, USA, and appropriate supporting personnel are being used to establish and conduct research under the title of the OECD LOFT Project. The LOFT facility is a 50 MW(t) pressurized water reactor system designed to provide information on reactor system response during abnormal events or accidents. It is particularly well suited for experiments and acquisition of data on the operational transients and multiple failure events, including fission product releases, that may occur in a commercial reactor. Its versatility provides an excellent means for assessing and developing techniques for managing accidents.

The OECD LOFT Project is managed by a Board with one member from each sponsoring country and organisation, namelly: Austria, Finland, Federal Republic of Germany, Italy, Japan, Sweden, Switzerland, United Kingdom, United States (NRC & DOE) and the US Electric Power Research Institute. The Project is expected to finish in 1986 at a cost of about US$ 90 million.

PRIMARY CIRCUIT INTEGRITY

The choice of subjects in this area for joint consideration is, of necessity, highly selective: work is now centred on questions relating to the primary circuit of light water reactors. Materials problems can occur because of physical processes in the hostile reactor environment and these can produce defects. Two complementary aspects are examined in the CSNI programme: fracture mechanics (theoretical analysis of the way flaws behave and their acceptability in service) and non-destructive methods of flaw detection (its precision and limitations). The CSNI programme is related specifically to methods, both theoretical and practical, under these two headings.

In fracture mechanics Member countries are moving slowly towards identifying common methods of calculation, particularly in the so-called elastoplastic regime, where stress is concentrated to such an extent - for example at the tip of a crack in a steel section - that it exceeds the yield point of the material which then becomes plastic. A similar common approach is being pursued in testing the fracture properties of steels.

In non-destructive examination (NDE) of the reactor primary circuit, ultrasound is used for locating and sizing flaws and evaluating their significance. The PISC-II* programme began operations in January 1982, with four heavy section test plates, each weighing about 10 tonnes and containing implanted defects being shipped to 14 countries for inspection by some 50 organisations. This programme was completed during the summer of 1984. Preliminary work is being done to outline a possible future project in the PISC framework on validation of NDE techniques on real service-induced defects.

In parallel with PISC projects, aspects of the reliability of NDE are being investigated and work began in 1983 on the theoretical modelling of ultrasonic examination, the problems of near-surface inspection and signal processing. A limited "round-robin" test programme was launched on samples of used, stainless steel reactor piping, a material which is especially difficult to inspect.

SOURCE TERM AND ENVIRONMENTAL CONSEQUENCES

Accident source terms

When considering the potential effects of a nuclear accident, the probable quantities, mix, and rate of emission of radioactive isotopes from the site of

---

* PISC-II is the second international Programme of Inspection of Steel Components; the first PISC programme was conducted from 1975 to 1979.

the damaged nuclear plant assume great importance. This "source term" was first described in the early 1960s when very little empirical information was available. The source term was then arbitrarily assumed to include a substantial proportion of the radioactive elements contained in the core of the reactor. The Three-Mile Island accident, however, showed that the highly active species of elements such as caesium, iodine and ruthenium produced in a core disruption are far more likely to react chemically and physically with their immediate surroundings than to migrate out of the containment and off the reactor site. This led to the hypothesis that the source term postulated for nuclear power plant accidents might be unduly high and the risk of public exposure to radioactivity from these accidents could have been overstated.

A number of countries are studying this question. Some preliminary results have been obtained; although a complete international evaluation in the framework of CSNI will not be available for a few years.

Air Cleaning in Accident Conditions

Air cleaning systems which service the auxiliary or secondary containment buildings in a nuclear power plant may have to play an important role in some types of accident. These systems have also crucial safety functions in various fuel cycle installations, particularly reprocessing plants. The way air cleaning systems behave in accident situations is not well known; work is continuing on developing more robust systems, and on devising international agreement on test methods for extreme conditions.

Accident Consequence Modelling

The aim of safety assessment is to evaluate the radiation dose to the public after radionuclides have been released into the environment. Different ecological exposure pathways are possible; there is the chance, for example, of direct external irradiation from radionuclides in the air, on or in the ground or clothes, for internal exposure after inhalation of contaminated air, or of ingestion of contaminated water or foodstuffs. An attempt has been made to identify source term characteristics which are important for analysing the offsite consequences of an accident; this work will provide essential guidance for selecting the source term issues that should be examined. A number of important exposure pathway parameters are being evaluated by a group of experts.

Environmental transport analysis to estimate the offsite consequences of reactor accidents is an important step in the Probabilistic Risk Assessment (PRA) of a nuclear plant. Although, in general, accident consequence models were developed in the 1970s for evaluating the aggregate risk of potential accidents at many reactors and sites, they are now being applied to examine the risks posed by reactors at specific sites and to provide guidance for planning and decision-making. Besides use in risk evaluation, other areas of important application include evaluation of alternative design features, emergency planning and response, reactor siting recommendations, and the development of acceptable risk criteria.

RISK ASSESSMENT

Probabilistic Risk Assessment (PRA) is finding increasing use as a tool in the design and regulation of nuclear power plants. During the last decade, comprehensive PRAs have been carried out in several countries, but it is clear that further development of methods and interpretation is needed if their full potential is to be realised. A critical review has begun of the analytical techniques that have been used in PRAs, their strengths and limitations, and how they could stand improvement.

A second study is surveying the uses that have been made of probabilistic arguments and risk assessments in safety-related decision-making. This review will show where the lessons learned from PRAs can confidently be put to use and where they must be used with caution if at all.

One potential application of accident sequence assessments is providing guidance to power plant operators faced with a severe accident. Information is being collected on how accidents involving serious core damage will evolve, how the remaining reactor systems can influence the destructive processes occurring, and what options are available to the operator either to stop the destructive processes, or minimise the consequences.

SEVERE ACCIDENTS

An accident exceeds the design basis when there is failure of structures, material, systems, etc., without which core cooling cannot be properly assured by normal means. This is considered as a severe accident whose seriousness depends on the degree of fuel damage and on the degree of loss of containment integrity.

A Senior Group of Experts on Severe Accidents was set up by CSNI at the end of 1980 with the aim of reviewing Member countries' current knowledge and positions regarding severe accidents and the expected response of existing safety systems.

The Group stressed that current designs of light water reactors, based on conservative assumptions, were in fact more capable of coping with severe accidents than the design basis accident assumptions would suggest. Consequently, the Group considered the most effective approach would be to make accident initiation less likely, as well as to reduce the probability of its propagating at every subsequent stage. It was concluded that the capability of a plant to function in conditions well beyond the design basis provided a margin of safety which should be exploited to maintain control over events and minimise the consequences to the public.

The most important area of concern is thus accident management throughout the whole sequence of a severe accident and highest priority should therefore be given to improving the ability of plant personnel to monitor, diagnose and influence the course of a severe accident from the earliest stages.

CRITICALITY PROGRAMS FOR THE TRANSPORT OF FISSILE MATERIALS

The nuclear fuel cycle involves handling, storing and transporting a range of fissile materials in many chemical and physical forms. Computer programs have been written to determine whether fissile material packages will remain sub-critical (i.e., that a divergent chain reaction will not occur) in various situations that may arise. A group of experts carried out an exercise in 1983 which compared several computer programs used in different countries to make such calculations.

This exercise showed that the programs can indeed give consistent and trust-worthy predictions for packages of fissile material in various configurations.

During 1984, the group of experts also began a study of the criticality of reactor fuel disintegrating in water. This will help in evaluating the hazards associated with a damaged reactor core or an accident involving spent fuel elements in water pools or shielded containers, and with the design of fuel reprocessing equipment. Similar studies are being started, in collaboration with the NEA Committee on Reactor Physics (NEACR), on the computer programs used for assessing shielding and heat transfer in spent fuel containers.

SOME NEA PUBLICATIONS
IN THE NUCLEAR SAFETY AREA

Safety of the Nuclear Fuel Cycle (A State-of-the-Art Report by a Group of Experts, 1981)

Sûreté du cycle du combustible nucléaire (Rapport sur l'état des connaissances établi par un Groupe d'Experts, 1981)

£6.60 US$16.50 F66,00

Critical Flow Modelling in Nuclear Safety (A State-of-the-Art Report by a Group of Experts, 1982)

La modélisation du débit critique et la sûreté nucléaire (Rapport sur l'état des connaissances établi par un Groupe d'Experts, 1982)

£6.60 US$13.00 F66,00

Ductile Fracture Test Methods (Proceedings of a Paris Workshop, 1982)

Méthodes d'essais en matière de rupture ductile (Compte rendu d'une réunion de travail de Paris, 1982)

£19.00 US$38.00 F190,00

International Comparison Study on Reactor Accident Consequence Modeling (Summary Report to CSNI by an NEA Group of Experts)

Comparaison internationale sur la modélisation des conséquences des accidents de réacteurs (Résumé d'un rapport au CSIN par un Groupe d'experts de l'AEN)

£9.50 US$ 19.00 F95,00

Nuclear Aerosols in Reactor Safety – Supplementary Report (Report to CSNI by an NEA Group of Experts) (in preparation)

Les aérosols nucléaires dans la sûreté des réacteurs – Rapport complémentaire (Rapport au CSIN par un Groupe d'experts de l'AEN) (en preparation)

Air Cleaning in Accident Situations (Report to CSNI by an NEA Group of Experts) (in preparation)

L'épuration des gaz en situations accidentelles (Rapport au CSIN par un Groupe d'experts de l'AEN) (en préparation)

67739