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SOME PRACTICAL IMPLICATIONS OF SOURCE TERM REASSESSMENT

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FOREWORD

Nuclear reactor accident source terms have been the subject of much national and international research. Over the past years, a significant amount of new information has been developed which attempts to provide a detailed, mechanistic technology for predicting the behaviour of radionuclides under severe reactor conditions. An important result of this research has been to demonstrate the critical dependence of accident source terms on the specific features of reactor designs and reactor accidents under consideration.

Although it has been recognized that it is doubtful whether there will ever be sufficient knowledge to resolve many of the uncertainities associated with the predictions of severe accidents source terms, the advances in science and technology and their application in various nuclear safety fields is appropriate at this stage of this technology development.

This report is based on a technical review of severe accidents source term affecting LWRs and does address a number of safety related issues where the reassessment effort might have importnant implications, mainly: accident management, emergency planning and response, and backfitting and new design.

EDITORIAL NOTE

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I. PURPOSE OF THE REPORT

This report provides a brief summary of the current knowledge of severe accident source terms and suggests how this knowledge might be applied to a number of specific aspects of reactor safety.

In preparing the report, consideration has been restricted to source term issues relating to light water reactors (LWRs). Consideration has also generally been restricted to the consequences of hypothetical severe accidents rather than their probability of occurrence, although it is recognized that, in the practical application of source term research, it is necessary to take account of probability as well as consequences.

The members of the Group considered that their main task was to advise on the extent to which the present understanding of severe accident source terms could be used to reassess a number of practical safety aspects for nuclear power plants. The specific areas identified were as follows:

(a) Exploration of the new insights that are available into the management of severe accidents (Section III);

(b) Investigating the impact of source term research on emergency planning and response (Section IV);

(c) Assessing the possibilities which exist in present reactor designs for preventing or mitigating the consequences of severe accidents and how these might be used effectively; exploring the need for backfitting and assessing the implications of source term research for future designs (Section V); and

(d) Improving the quantification of the radiological consequences of hypothetical severe accidents for probabilistic safety assessments (PSAs) and informing the public about the realistic risks associated with nuclear power plants (Section VI).

II. BACKGROUND

For the purposes of this report, the source term is defined as the quantity, timing and characteristics of the physical and chemical composition of the radioactive material that could be released to the environment during the course of a severe reactor accident. The range of accidents considered includes those postulated as design basis and siting basis accidents, as well as those more severe that could lead to potential containment failure. Accidents in the last class are now termed severe accidents.

Since the accident at Three Mile Island Unit 2 (TMI-2), increased attention has been given to the details of accidents that could take place at a nuclear power plant and that could result in radioactive releases to the environment in excess of acceptable levels. Because of the scale and complexity of the relevant postulated events, it is not possible to carry out direct experimental investigations to characterize them. Instead, these complex sequences have been modelled as a set of simpler component phenomena. Experimental programmes have been undertaken to characterize the results of occurrences of possible combinations of these phenomena in postulated events. The improved understanding of the various individual phenomena has been incorporated into analytical computer codes, thus synthesizing the individual phenomena into sequences of events representing hypothetical accidents.

Substantial research programmes have been undertaken by many organizations to develop the analytical codes, to provide the technical data required for code inputs and to demonstrate the understanding of elements of the overall model.

In the United States of America, principal efforts have been undertaken by Battelle Columbus Laboratories (BCL), Sandia National Laboratories (SNL) and the Oak Ridge National Laboratory (ORNL), by Brookhaven National Laboratory (BNL) under the sponsorship of the US Nuclear Regulatory Commission (USNRC), and by the Industry Degraded Core Rulemaking Programme (IDCOR), supported by the US nuclear industry. In these efforts, a number of similar accident sequences for a number of selected US plant designs have been analysed.

In Europe, analyses have been conducted for some accident sequences; for example, analyses have been performed in the Federal Republic of Germany for risk dominating sequences. In some of these cases, particularly for late containment failure, the calculated radioactive releases are very low and difficult to compare with those in the US work since the containment design differs greatly from US designs for the same type of plant. In the United Kingdom a considerable amount of research and analytical work has been carried out on the postulated severe accident source terms for the 1300 MW(e) pressurized water reactor (PWR) at Sizewell. French studies have analysed PWRs in France, and Swedish studies have analysed a BWR design.

In addition to the analyses listed above, many countries have sponsored experiments to derive supporting data and have developed analytical methods which can be used as part of the overall sequence analyses.

Recently, four separate reviews of the state of knowledge of source terms have been published in the United States. These are:

(a) Reassessment of the Technical Bases for Estimating Source Terms [1];

(b) Report to the American Physical Society [2];

(c) The report of the Industry Degraded Core Rulemaking Program (IDCOR) [3]; and

(d) The American Nuclear Society Report [4].

An assessment of these four reviews indicates a number of areas of general agreement which are summarized in the following together with some of the important topics which still remain open.

(1) Considerable progress has been made since 1975 in developing both a scientific basis and a calculating ability for predicting the source term. Nevertheless, there are still uncertainties associated with information on accident source terms. Further research is needed to explain factors such as natural circulation in the reactor vessel; core melt progression and hydrogen generation; the possible release of non-volatile radionuclides in core- concrete interactions; retention and revaporization of fission products in the reactor coolant systems; the distribution of aerosol particle sizes; the scrubbing efficiency of suppression pools; and containment failure mechanisms (pressure loads and failure modes).

(2) The risk to the public from the majority of severe nuclear accidents, especially risks presented by iodine, would be below that predicted in previous studies using earlier assumptions about the source term.

(3) For many accident sequences, the fission product source terms are likely to be lower than had been calculated in previous studies. This reduction can be attributed to three principal factors:

(i) the recognition that reactor containments are stronger than assumed earlier and therefore fail later, if at all;

(ii) inclusion in the modelling of previously neglected physical and chemical phenomena that would lead to the retention of fission products; and

(iii) inclusion of additional sites (auxiliary buildings, ice beds, suppression pools, etc.) that would retain radionuclides more efficiently than had previously been assumed.

(4) Caesium iodide (CsI) is expected to be the predominant form of iodine, as opposed to the assumption that iodine would be released from severely damaged fuel in elemental form. Iodine is now treated as being in the form of caesium iodide (CsI) rather than as elemental iodine. CsI is more likely to be present owing to thermodynamic considerations and the fact that caesium as a fission product is present in amounts roughly ten times as large as is iodine within the fuel pellets. Elemental iodine, however, may be present under certain environmental conditions.

(5) Tellurium is another important fission product whose release is strongly affected by its chemical environment. The retention of tellurium by the fuel until it is released during ex-vessel[1] oxidation of the cladding in certain accident sequences accounts for a calculated tellurium leakage

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1 That is, the release of radionuclides and the generation of aerosols in a severe reactor accident after core debris has penetrated the reactor vessel and entered the reactor containment. The processes considered include release during the expulsion of core debris from the reactor vessel, release caused by explosive interactions of molten core debris with coolant water in the containment, and release during attack by core debris on the structural concrete in the containment.

release fraction larger than the caesium and iodine leakage release fractions.

(6) Sequences and plant details are important in estimating the plant specific source term. Results obtained from the analysis of a variety of sequences for a number of different plants indicate that it would be very difficult to characterize source terms for LWRs by a single table of numerical values, or to transfer results from one plant to another without careful examination of plant specific details in system design, component design and selection, structural detail and actual system performance and reliability.

Analyses performed in the recent studies have not treated all types of reactor or all types of containments in equal detail. It is necessary, therefore, to continue the studies of different plant types and accident sequences.

(7) An area of high uncertainty at present is the interaction of molten core materials with the concrete floor after vessel melt-through (ex-vessel release). Continued experimentation as well as modelling efforts are expected to reduce considerably the level of uncertainty associated with the current estimates.

III. ACCIDENT MANAGEMENT

General considerations

The substantial improvement in understanding the fission product source term has been a result of both recent research and better methods of analysis. It has led to many changes in the way potential severe accidents are viewed. In particular, improved estimates of the source term from different conceivable accidents and from different ways of reacting to them make it possible to plan accident management more intelligently. It is possible to begin to reduce or eliminate conservatisms and non-physical assumptions that were previously made to compensate for insufficient detailed understanding of real effects. This permits nuclear plant designers, operators and regulators to concentrate their attention on practices that can provide real benefit in the realistic evaluation of plant behaviour and the use of systems.

One of the principal new insights is that much can be gained from measures to prolong the time sequence in the development of an accident. The engineered safety features in the plant provide numerous such possibilities to an operating staff that understands the plant well, as do many auxiliary systems provided for normal plant operation. It is important that the possibilities for use of these systems be understood.

The prolongation of time would permit intelligent operator intervention to use available existing systems to ensure continued cooling. Of course, the best solution would be operator intervention that prevented damage to the fuel and the consequent release of fission products from their confinement within the fuel cladding.

In order to optimize the management of an accident, it is important that the operating staff understand the characteristics of fission product behaviour and the effects on source terms and uses of systems that can be used to control the pressure, flow and water inventory of the primary and secondary coolant systems. Flexible use under abnormal operation of these systems and their components should be anticipated.

The types and numbers of such systems, as well as their designs, generally differ from one nuclear plant to another, so that the non-conventional use of plant systems (including non-safety related systems) to protect the facility and the public under abnormal conditions require exploration on a plant specific, or reference type specific, and detailed design bases.

It is important to develop plant specific concepts of how best to use the systems available to halt the progression of the accident by protection of the primary system boundary, the containment building and any additional systems and structures that augment the function of containment of fission products, such as filters, sprays, water pools and auxiliary buildings, in order to maximize the length and complexity of the pathways through which fission products would have to travel to escape to the environment. Some nuclear plants retain spent fuel in cooling pools in the reactor containment building. In these cases, methods of ensuring continued cooling of spent fuel under conditions of long term denial of access must also be developed. For some power plants such measures have already been taken into account in the present design or by implementing backfitting measures.

Accident progression

During that phase of an accident involving the core, in which the temperatures remain within the design basis limits, source term research indicates that nearly the entire core inventory of radionuclides remains within the fuel matrix. The use of all available plant systems, therefore, should be focused on preventing further increases in core temperatures.

If the accident has progressed so far as to overheat severely and damage the core, but long term coolability can be restored within the reactor vessel, emphasis should be placed on restoring the integrity of the reactor coolant system and/or the cooling capability. This action would also tend to maximize fission product retention in the primary system. Under these core conditions, a large fraction of the volatile fission products is likely to have escaped to the containment. Nevertheless, the prevention of further core degradation maximizes the likelihood that containment integrity will be maintained, thereby achieving retention of the volatile fission products in the containment. (Leakage from the containment must be anticipated, but is likely to be several orders of magnitude smaller than the containment inventory.)

If a substantial part of the fuel has reached a condition of liquefaction, to the extent that recovery of coolant injection cannot prevent penetration of the reactor vessel and subsequent core--concrete interaction, accident management efforts should focus on the reactor containment. Even though the majority of volatile fission products must be expected to have been released from the fuel under these conditions, and a release of the non-volatile inventory may result from the core--concrete interaction, a reduction of the airborne concentrations in the containment by one or more orders of magnitude can be expected if the containment integrity is maintained for a period of several hours to days.

Even if containment integrity cannot be maintained, the amounts of fission products released can be reduced by maximizing the retention of the fission products in auxiliary structures (secondary containments, auxiliary buildings, etc.) and attempting to route air flow paths through any available filtration systems.

Plant specific studies

These considerations suggest the need for a plant specific study of the mitigation of risk by operator action. For risk dominant sequences, the currently available improvements in thermal-hydraulic analytical methods and methods of calculating source terms should be exploited to calculate the effects of operator intervention to reduce damage and suppress the release of fission products. The new factors now available offer more realistic analytical methods that can guide actions more dependably, offering a real benefit.

The outcome should be a symptom based set of responses to provide a direct and rapid connection between the state of the system and the measures available to contain the situation. This would augment the ability to act in the light of a good physical understanding of the effects of operator actions on plant operations. A necessary prerequisite is training of senior operating staff that focuses on understanding of the plant, even under adverse conditions, and familiarity with the combined sets of symptoms, states and responses.

For this reason, responses to such events must be based on an assessment of the plant conditions (based on the symptoms on which information is available to the operator). To this end, sufficient preparation in terms of personnel, organization and technical expertise should be provided for.

Also inherent in the application of such an approach is the availability of wide range diagnostic instrumentation in the nuclear plant, so that a reliable link can be established between abnormal events and the information available to the staff to determine the state of the systems. The instrumentation should provide information concerning, for example, the following:

(a) The state of core cooling;
(b) The integrity of the reactor coolant system;
(c) The state of the heat sink;
(d) The integrity and state of closure of the containment;
(e) Important variables of the primary, secondary and containment systems;
(f) The existence, approximate magnitude and spread of any fission products that might be released; and
(g) Neutronics and criticality.

The nuclear industry has generally been moving to supply the required instrumentation and to use it in diagnosis and decision making by introducing operator aids such as, for example, the safety parameter display systems (SPDS), the critical safety function monitoring system (CSFMS) or intelligent computer aided process information systems. However, further efforts might be necessary for some power plants to increase the reliability of the necessary instrumentation in areas beyond the traditional design basis.

Some aspects of source term and thermal-hydraulic analysis are still not well enough understood to ensure the complete success of the process just described. For instance, uncertainty about the details of core--concrete interaction may imply that, as a conservative measure, means should be available to ensure an adequate cooling capability for water accumulating in the containment building recirculation sump.

It is recommended that plant specific analysis be conducted of the effects of operator intervention after the onset of risk dominant accidents, to protect the successive barriers to fission product release. This would lead to risk reduction factors for these accidents that take into account a more realistic understanding both of the source terms from accidents and of accident analysis.

The results would be used to relate the state of deterioration of the core, observed by means of the response of instruments in the control room, to measures to prevent further deterioration and to control the fission product source. These measures would use all available plant systems, which offer many possibilities for plant protection.

The analysis should be used to generate a symptom based response to the appearance of system deterioration. Incorporation of these responses into emergency operating instructions usefully supplement the intelligent application of decisions taken by technically informed plant management, some addition being required in any case since not all exigencies can be foreseen.

IV. EMERGENCY PLANNING AND RESPONSE

General considerations

Every reasonable effort is made in the design and operation of nuclear power plants to prevent accidents from happening and to limit the consequences of any accident that might occur. It is this which provides the main assurance of the safety of the public. However, it is also generally considered prudent to make contingency plans, for both on and off the site, for dealing with any accident that might occur at the plant. Such emergency planning varies somewhat from country to country but usually involves detailed preplanning of specific countermeasures to be used (such as sheltering, evacuation, control of contaminated foodstuffs) at certain distances (depending on the countermeasures), followed by less detailed planning for areas beyond those distances.

In preparing emergency plans, the operator and the relevant authorities usually have some limited number of 'reference accidents' in mind. The plan is drawn up in such a way that it can cope with the release associated with these reference accidents, although it may be capable of extension to deal with somewhat larger releases.

Although seldom if ever explicitly so stated, it is evident that the operations staff and regulatory authorities have taken account of the likelihood (or probability) of an accident sequence and its possible consequences in deciding whether or not the detailed emergency plan should cope with it. Thus it is evident that, for example, the likelihood of a large meteorite crashing directly onto a reactor building is considered sufficiently remote (also for the consequences expected) that no consideration is given to it in preparing emergency plans. By the same token, there is good reason to hope that continuing research on severe accidents will provide an adequate basis for responsible authorities to conclude that the likelihood of severe accidents at the extreme end of the range (such as the rupture of the containment by a steam explosion) is so small that such accidents need not be considered in emergency planning. This position has already been adopted in many countries.

At higher probabilities there is a range of accidents which may need to be considered in preparing emergency plans. For these accidents the radioactive releases would occur if the containment were breached in one of the following ways:

(a) excessive containment leakage;

(b) containment isolation failure;

(c) containment bypass ; and

(d) delayed containment failure due to overpressurization.

The new insights available from the source term research have important implications for all such accident sequences.

Longer time expected for containment failure

Research has shown that detailed emergency planning is not necessary in the event of a delayed containment failure due to overpressurization (for some reactor types) if the containment is initially intact and containment leakages are diverted to the environment through the exhaust air handling system via the filters and stacks.

One common conclusion which emerges from the various source term studies is that for most accident sequences the largest single factor affecting source terms is containment behaviour. The increase in the pressure is somewhat slower and containments are stronger than had previously been assumed, and therefore will fail, if at all, at later times than were assumed in the USNRC Reactor Safety Study [5], for instance. This increased time to containment failure has three important implications:

(a) It gives the plant operators more time in which to try to recover the plant and prevent the eventual failure of the containment;

(b) It gives more time for natural, passive mitigating processes to take effect. Several mechanisms will lead to the deposition of aerosols onto surfaces both within the primary system and within the containment, and the amount of radioactivity released to the environment will therefore be less than was previously assumed; and

(c) Delayed containment failure also gives more time in which to set in motion the various parts of the emergency plan, both on and off the site.

Containment bypass

Another important result from source term research is that the prediction of the source term when the containment has not been isolated or has been bypassed is very sensitive to the details of the failure. The containment bypass[2] sequences are specific to each reactor design and need to be identified for each plant. When the sequences have been identified, their probabilities and consequences can often be reduced by quite simple steps. However, it seems unlikely that the probability of containment bypass accidents can ever be reduced to negligible levels, and so the consequences of such events will continue to require consideration in the formulation of emergency plans. Nevertheless, it is generally accepted that the radiological consequences of containment bypass accidents would be smaller than had been predicted earlier (e.g. in the Reactor Safety Study [5]). This is because it is argued that:

(a) Containment bypass releases would sometimes occur via a water or steam pathway, so reducing the amount of non-gaseous fission products released; and

(b) The fission products would most probably be trapped, to a greater extent than was formerly assumed, in various parts of the release pathway.

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2 Interfacing systems loss of coolant accident (LOCA, Sequence V); this sequence was the largest individual contributor to risk identified in the USNRC Reactor Safety Study [5]. The interfacing LOCA is of interest because the pathways for release bypass the protection normally provided by the containment building.

Different types of nuclides released

A third major conclusion from the source term research, which has important implications for emergency planning and response, is that the inclusion of previously neglected physical and chemical phenomena should, in most cases, lead to significant retention of fission products within the primary system and the containment, and that the range of released radionuclides will be different from that previously assumed. The potential releases of the noble gases krypton and xenon are not considered to differ significantly from those calculated in the Reactor Safety Study [5], except that radioactive decay will reduce their radioactivity if containment failure is delayed. The American Physical Society study noted that the reduction is by a factor of five between a two-hour release and a 24-hour release.

However, the chemical forms of some important fission products favour retention rather than release.

Caesium iodide (CsI) is expected to be the predominant form of iodine released and iodine is now treated as being in the form of caesium iodide (CsI) rather than as elemental iodine.

CsI is more likely to be present because of thermodynamic factors (its low vapour pressure and higher solubility in water) and the fact that fission product caesium is present in amounts roughly ten times as large as iodine within the fuel pellets. The caesium not reacted with iodine is thought to react with steam to form caesium hydroxide (CsOH). Elemental iodine may be present under certain environmental conditions, however.

Elemental iodine has been observed experimentally as a consequence of hydrogen combustion in the presence of aerosols containing CsI; elemental iodine and hydrogen iodide are also believed to be formed in certain types of reactor vessels (in BWRs) as a consequence of reactions with boron carbide control materials.

Iodine's propensity to react with organic materials to form volatile organic compounds will probably define a limit for iodine release, since the removal processes that would lower airborne concentrations of caesium iodide and other aerosols are not effective for organic forms of iodine. The process of formation of organic iodine and the range of this release are uncertain.

Tellurium is another important fission product that is strongly affected by its chemical environment. It is an extremely reactive material, forming compounds with unoxidized Zircaloy, and remains with the core debris as long as most of the cladding remains unoxidized. The retention of tellurium by the fuel until it is released during ex-vessel oxidization of the cladding for certain accident sequences accounts for a calculated tellurium leakage release fraction larger than the caesium and iodine leakage release fractions.

Thus there are strong indications from recent research that the source terms for several of the more important radioisotopes (particularly I, Cs and Te) in a number of important sequences are smaller than had been stated in the Reactor Safety Study [5], although research studies draw attention to some calculations which indicate the possibility that certain phenomena (e.g. the release of non-volatile nuclides during core- concrete interaction) might increase some radionuclide source terms for certain sequences to values above those in the Reactor Safety Study [5].

In spite of these uncertainties, it would seem appropriate for those responsible to reassess the current focus on radioiodine of emergency plans for LWR stations. In particular, this could have important implications for the emergency action levels that are used in the emergency plan, for the types of survey instruments that are used to monitor the radioactive plume both on and off the site, and for the use of potassium iodide or iodate as a medical prophylactic to limit radioactive iodine uptake in the human thyroid gland. One should remember, however, that reliable on-site monitoring in the event of a large leak in the containment and releases via the auxiliary building is usually not feasible.

It would also appear appropriate to review the relative emphasis placed on all emergency protective measures, including sheltering, evacuation or respiratory protection, in the light of the foregoing finding, and perhaps review the distances over which these are implemented. The facilities and procedures for limiting the exposure of operators and on-site emergency teams (such as emergency control room ventilation systems) may also need to be reassessed.

Some experts conclude from present source term research that another aspect of emergency planning which may require re-evaluation concerns the criteria assumed for controlling contaminated foodstuffs. If radioiodine is no longer expected to be released in significant amounts in most sequences it may no longer be appropriate to emphasize the dose saving which would result from the control of milk contaminated with iodine, for instance. It may also be necessary to consider whether the transfer of fission products through environmental pathways and in biological systems needs to be reassessed in the light of our improved understanding of the chemical and physical forms of the released radionuclides.

Other experts deny such far reaching conclusions. In spite of this controversy it seems appropriate to re-evaluate emergency planning — if not effected already -- at least for accident sequences where the release is delayed for a significant period of time. This requires a flexible response in emergency measures according to the actual circumstances pertaining.

The other relatively well established finding of source term research, namely that in most sequences the release from the containment is likely to be delayed for a significant period of time, could have important implications for the preparation and implemantation of emergency plans. Apart from allowing more time for recovery actions on the site, the delay before release to the environment should allow emergency services more time to set necessary actions in motion.

In conclusion, it is not expected that the findings of source term research will lead to a change in the requirement to have emergency plans, which are basically needed to provide assurance that all possible steps have been taken to mitigate the consequences of any accident that might occur.

Nevertheless, it is considered that, although there are still a number of matters to be resolved, the present understanding of severe accident source terms is sufficient to permit a reassessment of the basis of current emergency plans for LWR stations. Such a reassessment should ensure that the emergency plans not only reflect the current understanding of the likely mixture of fission products released in the majority of sequences, and their physical and chemical form, but also the expectation that containment failure, if it occurs at all, will usually be delayed for a significant time. For some reactor types, emergency measures are not absolutely necessary if the containment does not fail and if excessive containment leakages are avoided. More attention should be given, however, to the implications of possible containment bypass, preopening and isolation failure accidents for emergency planning.

V. BACKFITTING AND NEW DESIGN

Existing design philosophy

Current plant designs include engineered safety features (ESFs) for the purpose of mitigating the consequences of postulated accidents[3]. These systems would perform the desired function by containing contaminants, by reducing pressure driving forces or by removing contaminants from fluids which could leak to the outside environment.

The engineered safety feature most important to the control and mitigation of fission product releases is the reactor containment. The reactor containment, in conjunction with related containment systems (e.g. enclosures or auxiliary buildings, penetrations, etc.), is the outermost of several sequential barriers serving as protection against uncontrolled releases of radioactive material from the reactor core.

In addition, nuclear power plants designed in various Member States include a number of safety systems designed to function in conjunction with the reactor containment by removing fission products from the containment atmosphere (spray additive systems, recirculation filter systems), by reducing containment pressure and airborne contaminants (containment spray, pressure suppression pools, fan coolers, ice condenser systems), and by filtering the air likely leakage pathways from the containment (standby gas treatment systems, auxiliary building filtration systems).

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3 Engineered safety features consist principally of a number of systems designed to prevent the accidental release of fission products (e.g. by providing a core cooling capability). The present discussion does not include preventive type ESFs since source term research has no implications for them.

The accident at the Three Mile Island (TMI) nuclear plant on 28 March 1979 illustrates the validity, as well as the limitations of this approach to ESF design. On the one hand, the ESF design concepts (low leakage containment, containment spray additives for pH control) accommodated accident conditions (in particular core damage conditions) beyond the design basis. On the other hand, transfer of highly radioactive liquid from the containment sump to the auxiliary building, and lack of adequate capability to deal with high levels of radioactivity in the auxiliary and fuel handling buildings, illustrate the omission of the safety system designers to anticipate accident conditions beyond the design basis.

The existing design approach is to postulate a non-mechanistic design basis accident (DBA) as an aid in the specification of design parameters for these systems. In order to simplify the consideration of the very complex situations, DBA conditions postulated in this approach usually neglect any beneficial effects (e.g. fission product attenuation) arising from natural phenomena.

The resulting design is intended to be applicable to a wide range of conditions, including, but not necessarily limited to, the DBA conditions, and, as a result of the safety margin incorporated into the DBA conditions, is expected to extend ESF capabilities to conditions somewhat more severe than those in the DBA.

With the improved understanding of fission product behaviour resulting from the results of source term research, it is appropriate to re examine the effectiveness of past ESF designs. A qualitative review of this type [6] indicated that, as a result of the general approach to the design of ESFs, the effectiveness of ESF systems generally extends to conditions considerably beyond the DBA conditions owing to the large safety margins incorporated into their designs. The evaluation of expected conditions for severe accidents, however, may suggest some design modifications to alter their operating capabilities.

Source term research affecting ESF design

The extensive research on fission product release and transport is currently producing quantitative results applicable to ESF design. New insights into fission product attenuation by natural[4] mechanisms indicate that these processes would be more effective than had previously been assumed.

The physical and chemical state of the radionuclides, for example, is expected to be altered by interactions with non-radioactive bulk materials (water, structural materials, control rod materials, fuel cladding, etc.) within the primary pressure boundary. As the volatile fission products (i.e. Xe, Kr, Cs, Rb, I, Sb, Te) are carried in an atmosphere of steam and hydrogen through the primary system, some might condense on stainless steel or other surfaces, or on aerosols formed from vapours of control rod or structural materials. These species might be partially retained in the reactor structure without ever reaching the containment. (The inert gases Xe and Kr, however, would not be affected.) In the lower temperature environment of the containment atmosphere, the physical form of non-inert fission products, their oxides and other chemical species is expected to be primarily liquid or solid aerosols. Aerosol processes, therefore, will govern the mobility of most fission product species. Volatile species of

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4 The term 'natural' is used to refer to processes occurring during the accident which were not specifically considered in the design of the safety system.

iodine may be formed by radiochemical processes, but these are expected to be relatively small fractions of the total iodine inventory.

Within the containment, aerosol processes such as agglomeration, settling, diffusiophoretic and thermophoretic deposition on cooler containment surfaces, and interaction with steam and suspended water are expected to contribute to fission product attenuation. Reductions of the airborne concentration by an order of magnitude every four to six hours for the first two or three days have been estimated in the absence of active ESF (i.e. filters or sprays) operation. However, complete reduction of airborne concentrations below a certain limit is not expected. The aerosol behaviour in the containment is likely to be dominated by non-radioactive aerosols. For accident sequences resulting in vessel melt-through and subsequent core- concrete interaction, it has been estimated that as much as several tonnes of non-radioactive material would be injected into the containment atmosphere. The amount is very uncertain, however.

These phenomena could significantly affect the performance of engineered safety features. For example, the understanding of the required performance of the containment has changed significantly as a result of the improved understanding of the natural attenuation processes. Whereas the existing concept for containment design based on the retention of volatile iodine forms requires an essentially leaktight structure for an extended period of time (i.e. one month) after an accident, new insights concerning fission product behaviour indicate that the effectiveness of aerosol processes can be assured by maintaining reasonable integrity for a period of approximately one day. One may conclude from this that an effective use of resources is to assure that large pre-openings in the containment do not exist and that containment isolation failures are avoided with a high degree of reliability, rather than the verification of extremely small leakage rates over long periods of time.

Plant design modifications

It should be noted that the insights gained from new source term research do not invalidate current plant designs. Rather, the additional information concerning fission product behaviour indicates that even the safety systems currently installed that are directed at protecting the nuclear power plant retain at least some of their effectiveness for accidents beyond the design basis. Current research has shown some capability of many ESFs to protect the public even if accidents occur that are beyond the design basis. Nevertheless, severe accident conditions can be postulated for which the existing systems would be less effective than desired. The question arises, therefore, whether there is a need for backfitting existing plants with alternative ESF systems, or for modifying existing systems.

Several options exist:

(a) To maintain the status quo, except for the correction of significant design errors, if any are identified;

(b) To perform quantitative evaluation of existing designs and to optimize the installed ESFs by minor adjustments, particularly as ageing of plant requires the replacement of equipment; and

(c) To design and backfit entirely new engineered safety features, such as vented filtered containments. To date however, no single major 'new' system has been demonstrated to be effective for the mitigation of the complete range of risk dominant severe accidents.

Recommendations for new plants

Where an opportunity exists to incorporate fully the new source term information, such as in the design of new plants, the following approach would be recommended:

(a) To use PSA to evaluate current designs and, if cost effective, to select design features most efficient for risk-dominant accidents. The use of detailed PSA methods will be most valuable in application to standard plant designs. To ensure the optimum value of this use of PSA, standardization should include not only the nuclear steam supply system, but secondary and auxiliary systems, as well as structures; and

(b) The use of the current research results on the effectiveness of natural processes by emphasizing passive systems such as containment and scrubbing by water pools. Re-examination of systems to prevent the bypassing of these processes (e.g. containment bypass or containment isolation failure) may be necessary.

VI. IMPROVED KNOWLEDGE OF RISK: PROBABILISTIC SAFETY ANALYSIS [5]

Full scope probabilistic safety assessments (PSAs) use detailed source terms to estimate the transport of the radionuclides in the environment, the resulting doses to the affected population, and other consequences such as land contamination. The results of these accident consequence calculations are assessed with their respective frequencies.

Although many experts in the field of reactor safety analysis agree that current source term values for given occurrence frequencies used in PSA studies are too large, there is much uncertainty over the extent to which these estimates could be reduced with rigorous technical justification. This is a reflection of the complexity of the physical phenomena and the uncertainties in modelling the in-vessel subprocesses of core meltdown. It is also due to the inadequate ability to predict the structural response of the containment during accidents due to phenomena such as steam explosions or hydrogen detonations, and their frequencies, if such phenomena of sufficient magnitude to cause the containment to fail cannot be absolutely excluded for specific plant designs on theoretical grounds.

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5 Risk is defined as the frequency of occurrence for a postulated sequence multiplied by its off-site consequences; the process of estimating the risk for a plant includes identification of all accident sequences and their frequencies, possible containment failure modes (or continued containment integrity), calculation of source terms, calculation of off-site consequences for each of the source terms, and calculation of an estimate of risk. The sum of all risk estimates that cover all the dominant sequences would provide a risk estimate for the plant.

The Probabilistic Risk Analysis Procedure Guide [7] gives an example of the effect on predicted early fatalities if the source terms used for given frequencies in the Reactor Safety Study [5] were reduced by factors of five or ten, and also the impact of the same reductions on early injuries, latent cancer fatalities and areas interdicted for ten years or more. Some authors have argued strongly that such reductions in source terms are justifiable on the basis of existing evidence. Others argue that the case for such reductions is not yet proved.

Uncertainties in techniques of probabilistic safety analysis (PSA) lead to considerable controversy concerning the validity, interpretation and use of PSA results, and they limit the wider implementation of quantitative safety goals into the current licensing and regulatory processes in Member States.

It should, however, be remembered that PSA is a methodology dealing rigorously with uncertainties. Also, PSA is making an explicit statement about uncertainties, whereas these uncertainties are irretrievably embedded in deterministic alternatives. Decision makers have had to make decisions under uncertainty in the past, and will have to continue to make such decisions in the future.

More precise and 'best estimate' source terms, based on greater knowledge which should be used in PSA studies, will improve the accuracy of evaluation of fission product transport, deposition and dispersion, and health effects and their probabilities. This should allow more realistic estimates to be made of the possible risks from severe nuclear power plant accidents, and so give a more reliable and confident basis for informing the public about the risks from nuclear power generation and for checking the extent to which such risks run counter to any relevant safety goals or objectives. More important, the new source term information should increase the quality of information for a well informed, and hence more refined, decision making process in the practical application of PSA for risk reduction efforts and a well balanced safety concept.

SUMMARY

It is recognized that information on source terms is incomplete and has uncertainties associated with it.

Large source terms could occur only through a combination of unfavourable circumstances whose potential interactions become increasingly difficult to quantify. It has been recognized by some experts that it is doubtful whether there will ever be sufficient knowledge to provide complete and definitive information with no uncertainties for such low probability accidents. Nevertheless, there have been sufficient advances in knowledge of fission product release and behaviour following severe accidents that the practical application of this information in various nuclear safety fields is now appropriate.

The activities of the IAEA in this area should take these facts into account and should be directed towards the following applications of the reassessment of source term research.

(1) The new information available on source terms should be applied to the management of severe accidents.

Source terms were found to depend strongly on details of plant design and construction; therefore plant specific details must be taken into account in the identification and selection of the most suitable actions to optimize effective ways to prevent the release of fission products.
It is therefore recommended that plant specific analyses be conducted on the effects of operator actions after the onset of risk dominating accidents, to protect the successive barriers to the release of fission products. The

analysis should be used to generate a symptom based response to the observation of system deterioration. The existing variation in accident management procedures should be checked and assessed in the light of the reassessment of source terms for severe accidents.

(2) The effect of the results of the source term reassessment on emergency planning procedures and responses should be investigated.

The present understanding of severe accident source terms permits a reassessment of the technical basis of existing emergency plans; in particular, the likelihood of the maintenance of containment integrity for substantial periods of time, the expected mixture of radionuclides and their physical and chemical forms. For most accident sequences analysed, the most important single factor affecting source terms is the containment behaviour. It is not expected that the findings of source term research will obviate the need for off-site emergency plans as a measure for the protection of the public from highly unlikely but potentially severe accidents. However, the findings could lead to an easing of the urgency, a reduction in the areas involved and the modification of the measures foreseen for emergency planning.

(3) The possibilities with present reactor designs for averting or mitigating the consequences of severe accidents should be assessed.

With the improved understanding of fission product behaviour gained by source term research, it is appropriate to re examine the effectiveness of existing engineering safety features (ESFs) and other operational systems not credited in licensing submissions. The preliminary conclusion is that the effectiveness of existing ESF systems in combination with other existing operational capabilities (including non-safety related and auxiliary systems) is generally maintained in conditions considerably more severe than the design basis accident (DBA) conditions.

(4) The need for backfitting should be investigated and the implications for future designs should be explored.

Any attempt to examine a need for additional safety provisions (backfitting) should be assessed in the light of existing safety principles, translated into regulatory requirements, to ensure that public safety objectives are being met with the existing design concept. The evaluation process must take into account the totality of current knowledge of the source term (especially a realistic assessment of ESFs and other system capabilities beyond DBA conditions and containment behaviour), as well as probabilities of the dominant accident sequences for individual risk and the cost effectiveness of any additional provisions.

The insights from current source term reassessments should be incorporated into the design philosophy for new nuclear power plants, with specific attention being paid to severe accident sequences having higher probabilities of occurrence attributed, and to the effectiveness of natural processes in retaining fission products.

(5) The evaluation of that part of probabilistic safety assessment (PSA) relating to radiological consequences should be improved.

The re-evaluation of the frequencies of occurrence of various accident sequences and an improved knowledge of source terms (plume characteristics and release fractions) and containment failure modes will help in the evaluation of radiological consequences. This should allow more realistic estimates to be made of the possible risks due to severe nuclear power accidents, and thereby provide a more reliable basis for informing the public about the risks associated with nuclear power generation and for checking what bearing such risks have on any relevant safety goals or objectives.

(6) The availability of computational capabilities relating to the source term.

Since the determination of severe accident source terms must rely heavily on the use of complex computer codes, it is essential that the methodology for predicting the source term be available to all interested Member States. It has been recommended that the IAEA should provide a computational capability relating to the source term to assist Member States in performing analyses of this type.

At first, the capability should concentrate on process oriented methods of application such as the transport of aerosols and their behaviour in the containment, retention of fission products and aerosols in pools of water, effectiveness of ESF systems, etc. This capability should be seen as an introduction to the use of the source term methodology being developed for performing calculations for practical purposes. As concluded in the recent source term reassessment study by the United States Nuclear Regulatory Commission [1], the analytical procedure is complex and draws on several scientific disciplines. Successful application of the analytical procedure requires a thorough understanding of the problem to be solved, including the plant characteristics, the description of the accident sequence and the purpose of the analysis.

REFERENCES

[1] UNITED STATES NUCLEAR REGULATORY COMMISSION, Reassessment of the Technical Bases for Estimating Source Terms, NUREG-0956, Washington, DC (1986).

[2] REPORT TO THE AMERICAN PHYSICAL SOCIETY OF THE GROUP ON RADIONUCLIDE RELEASE FROM SEVERE ACCIDENTS AT NUCLEAR POWER PLANTS, Mod. Phys., 57, 3 (1985).

[3] THE REPORT OF THE INDUSTRY DEGRADED CORE RULEMAKING PROGRAM (IDCOR), Nuclear Power Plant Response to Severe Accidents (1984).

[4] AMERICAN NUCLEAR SOCIETY, The Report of the Special Committee on Source Terms, ANS, New York City, NY (1984).

[5] UNITED STATES NUCLEAR REGULATORY COMMISSION, Reactor Safety Study: An Assessment of Accident Risks in the US Commercial Nuclear Power Plants, WASH-1400 (NUREG-75/014), Washington, DC (1975).

[6] UNITED STATES NUCLEAR REGULATORY COMMISSION, Technical Bases for Estimating Fission Product Behaviour During LWR Accidents, NUREG-0772, Washington, DC (1981).

[7] UNITED STATES NUCLEAR REGULATORY COMMISSION, Probabilistic Risk Assessment Procedures Guide, NUREG/CR-2300, Washington, DC (1983).

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