Nuclear Safety ISBN 978-92-64-99008-1

CSNI Technical Opinion Papers

No. 9

Level-2 PSA for Nuclear Power Plants

NEA No. 5352

NUCLEAR ENERGY AGENCY
ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

ORGANISATION FOR ECONOMIC CO-OPERATION AND DEVELOPMENT

The OECD is a unique forum where the governments of 30 democracies work together to address the economic, social and environmental challenges of globalisation. The OECD is also at the forefront of efforts to understand and to help governments respond to new developments and concerns, such as corporate governance, the information economy and the challenges of an ageing population. The Organisation provides a setting where governments can compare policy experiences, seek answers to common problems, identify good practice and work to co-ordinate domestic and international policies.

The OECD member countries are: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Korea, Luxembourg, Mexico, the Netherlands, New Zealand, Norway, Poland, Portugal, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities takes part in the work of the OECD.

OECD Publishing disseminates widely the results of the Organisation's statistics gathering and research on economic, social and environmental issues, as well as the conventions, guidelines and standards agreed by its members.

This work is published on the responsibility of the Secretary-General of the OECD. The opinions expressed and arguments employed herein do not necessarily reflect the official views of the Organisation or of the governments of its member countries.

NUCLEAR ENERGY AGENCY

The OECD Nuclear Energy Agency (NEA) was established on 1st February 1958 under the name of the OEEC European Nuclear Energy Agency. It received its present designation on 20th April 1972, when Japan became its first non-European full member. NEA membership today consists of 28 OECD member countries: Australia, Austria, Belgium, Canada, the Czech Republic, Denmark, Finland, France, Germany, Greece, Hungary, Iceland, Ireland, Italy, Japan, Luxembourg, Mexico, the Netherlands, Norway, Portugal, the Republic of Korea, the Slovak Republic, Spain, Sweden, Switzerland, Turkey, the United Kingdom and the United States. The Commission of the European Communities also takes part in the work of the Agency.

The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Cover credits: GPU Nuclear Inc. (photo) and Pekka Pyy (drawing).

FOREWORD

The main mission of the NEA/CSNI* Working Group on Risk Assessment (WGRisk) is to advance the understanding and utilisation of probabilistic safety analysis (PSA) in ensuring the continued safety of nuclear installations and in improving the effectiveness of regulatory practices in NEA member countries.

In pursuing this goal, the working group examines the different methodologies for identifying contributors to risk and assessing their importance, while continuing to focus on the more mature PSA methodologies for level-1, level-2, internal and external events, and shutdown conditions. It also considers the applicability and maturity of PSA methods for addressing evolving issues such as human reliability, software reliability and ageing issues, as appropriate.

Technical opinion papers (TOPs) are considered to be one of the most important products produced by the WGRisk, and as such are produced in conjunction with the issuance of any new report, the completion of a workshop or following in-depth discussions. Recent TOPs have addressed PSA-based event analysis, living PSA and the development and use of risk monitors at nuclear power plants.

This TOP took as a starting point the work carried out to produce the report on level-2 PSA methodology and severe accident management, published in 1997 [NEA/CSNI/R(97)11]. This report provided a very detailed account of the state of the art at that time and gave insights into the methodologies that have been developed, the results of the analyses carried out for different types of nuclear power plants and the accident management strategies that have been devised. This TOP has also taken into account the information presented at the international workshops organised by the NEA on level-2 PSA held in Cologne in 2004 and on the evaluation of uncertainties in relation to severe accidents and level-2 PSA held in Aix-en-Provence in 2005. The next stage in the work to be

---

* CSNI: NEA Committee on the Safety of Nuclear Installations.

The NEA Secretariat wishes to express particular thanks to Dr. Charles Shepherd, who was the task leader for the WGRisk, and to Dr. Ming Ang. They were the main authors of the paper and have contributed their valuable time and considerable knowledge to its preparation.

Thanks are also extended to Mark Leonard, Bernard Chaumont and Paul Boneham who have helped with the preparation of the paper, to the members of the WGRisk and GAMA who have reviewed and commented on drafts of the paper and to Mike Evans who carried out a review of the final draft.

TABLE OF CONTENTS

LEVEL-2 PSA FOR NUCLEAR POWER PLANTS

Introduction

Probabilistic safety analysis (PSA) of a nuclear power plant provides a comprehensive, structured approach to identifying accident scenarios and deriving numerical estimates of the risk to members of the public from the operation of the plant. The insights gained from the PSA are used along with those from the deterministic analysis in the decision-making process on safety issues for the plant.

PSAs are normally performed at three levels as follows:

- Level-1 PSA which starts from an initiating event or an internal or external hazard that challenges the safe operation of the plant and identifies the combinations of failures of the safety systems that can lead to core damage. This provides an estimate of the frequency of core damage and gives insights into the strengths and weaknesses of the safety systems and the emergency procedures provided to prevent core damage.
- Level-2 PSA which models the phenomena that could occur following the onset of core damage that have the potential to challenge the containment integrity and lead to a release of radioactive material to the environment. The analysis considers the effectiveness of the design and the severe accident management measures that can mitigate the effects of core damage, and provides an estimate of the frequency and magnitude of a release of radioactive material to the environment.
- Level-3 PSA which models the consequences of a release of radioactive material to the environment and provides an estimate of the public health and other societal risks such as the contamination of land or food.

Level-1 PSAs have now been carried out for most of the nuclear power plants worldwide. However, in recent years, the emerging standard has been for level-2 PSAs to be carried out for all types of nuclear power plants. To date, level-3 PSAs have been carried out for relatively few plants.

The level-2 PSA provides a structured assessment of the possible accident sequences that could occur following core damage and gives insights into which of the phenomena that could arise have the greatest potential to lead to containment failure or bypass resulting in a release of radioactive material to the environment.

The results of the level-2 PSA can be used to determine if sufficient provisions have been made to terminate core damage at an early stage and hence prevent failure of the reactor pressure vessel and to mitigate the effects of the severe accident should the core damage progression continue. This gives insights into the level of robustness of the containment in providing protection for severe accidents and the adequacy of the accident management systems (such as the hydrogen mixing/recombining/igniting systems, containment spray systems and containment filtered-venting systems) in protecting the containment and preventing a large release of radioactive material to the environment. In addition, the level-2 PSAs are being used to identify any additional accident management measures that could be carried out to further mitigate the effects of a molten core (both inside and outside the reactor pressure vessel).

The results of the level-2 PSA can be compared with regulatory criteria such as the large early release frequency (LERF) (where such criteria have been defined) to provide an overall indication of the robustness of the safety systems and the containment in mitigating the effects of a severe accident. They can also be used to provide an input to the civil authorities that can be used as a basis for off-site emergency planning.

The level-2 PSA methodology is now seen as mature and is an essential part of the safety analysis that is carried out for nuclear power plants worldwide.

Background

A large number of level-2 PSAs have been carried out for a variety of nuclear power plant designs and, as a result of this, the overall approach is relatively well developed. The state of the art in performing level-2 PSAs up to 1997 has been described in [1] based on the existing studies for 19 pressurised water reactors and boiling water reactors. The report provides an overview of the methodologies used for the level-2 PSAs and describes how the insights obtained have been used to develop severe accident management strategies. The report also provides a comparison of key features adopted in the US NRC NUREG-1150 study [2], examples of US individual plant examination (IPE) studies and the other PSAs that had been carried out at that time. Further developments in more recent and ongoing level-2 PSAs are described in [3]. A procedure for performing a level-2 PSA and guidance for carrying out a regulatory review of a level-2 PSA has been produced by IAEA – see [4] and [5] respectively. This Technical Opinion Paper gives an introduction to level-2 PSA.

The concept of a level-2 PSA was introduced in WASH 1400 – the Reactor Safety Study published in 1975. Since then, significant progress has been made in the development of the methodology and the performance of plant-specific studies worldwide. The key stages in the development of the methodology and the insights that have been derived regarding plant vulnerability to severe accidents for light water reactor designs are described below.

The analysis carried out for the Zion and Indian Point nuclear power plants (1981) involved a more structured and expanded approach compared to the WASH 1400 methodology and made use of the early generation of the severe accident analysis code MARCH to support the probabilistic quantification of the severe accident sequences. The PSAs that were performed in the USA and Europe in the early 80s were largely based on this methodology. However, an independent study, the German Risk Study using Biblis B as the reference plant, was also carried out at this time. This was conducted in two phases and led to research and development activities being carried out that were aimed at providing a better understanding of severe accident phenomena – for example, the early BETA experiments were performed to support the validation of the WECHSL code for the analysis of core-concrete interactions.

The level-2 PSA methodology was significantly expanded in the NUREG-1150 study on severe accident risks based on five reference plants in the USA [2]. This study made use of very large containment event trees (referred to as accident progression event trees – APETs) and included an integrated uncertainty analysis for the entire PSA. The source term code package (STCP) was adopted for the severe accident analysis and other codes were developed specifically to support the study including a simplified code for source term generation (XSOR) and a code for quantifying the containment event trees (EVNTRE). The study coincided with intense research activities into the phenomena that would be expected to occur during a severe accident including high pressure melt ejection, direct containment heating, induced creep rupture failure of the reactor cooling system piping or steam generator tubes, and fission product release and transport behaviour.

In 1988 the US NRC issued Generic Letter GL 88-20 relating to individual plant examination for severe accident vulnerabilities and this led to level-2 PSAs being performed for all plants in the USA. There were significant differences in the approaches used in these assessments and in their level of detail. Some assessments made use of the EPRI NSAC-159 methodology (which is similar to the traditional level-1 PSA approach in that it uses a combination of event trees and fault trees) and the Modular Accident Analysis

Program (MAAP) code while others made use of the appropriate reference plant study in NUREG-1150. NUREG-1560, which provides observations on the results of the IPE programme, states that “in many of the IPEs, the containment performance analysis and the source term calculations are more simplified than one would expect to find in a current PRA” .

The findings from the level-2 PSA and deterministic code calculations have been used to extend the emergency operating procedures (EOP) from the design basis area to the severe accident domain and to identify the severe accident management (SAM) measures for mitigating the effects of a severe accident. The accident at Unit 2 of Three Mile Island (TMI-2) in 1979 and the insights from severe accident analyses and level-2 PSAs have shown that there is the potential to control plant states even beyond design limits and the SAM initiative is aimed at providing a structured capability in severe accident mitigation, if successive preventive measures have failed to restore core cooling. This structured approach requires an understanding of the plant vulnerability that would typically be achieved from a plant-specific level-2 PSA.

Aims of the level-2 PSA

The level-2 PSA follows on from the level-1 PSA and provides an integrated analysis that takes account of plant specific features to determine how the fault sequences that have occurred leading to core damage would progress to challenge the containment and lead to a release of radioactive material to the environment. The main aims of the level-2 PSA are:

- To gain insights into how severe accidents progress and identify plant specific vulnerabilities.
- To determine how severe accidents challenge the containment and identify major containment failure mechanisms.
- To estimate the quantities of radioactive material that would be released to the environment for different types of accident sequences.
- To determine the overall frequency of a large release of radioactivity to the environment.
- To evaluate the impacts of various uncertainties, including assumptions relating to phenomena, systems and modelling, on the magnitude and frequency of the release.
- To provide a basis for the identification of plant specific severe accident management measures and determine their effectiveness.

- To provide a basis for the identification and prioritisation of research activities to address the areas of uncertainty that have the highest risk significance.
- To provide an input into the development of off-site emergency plans.
- To provide an input into the decision making model for the timely execution of the off-site emergency actions.

Hence, there is a need to produce an integrated analysis that is based on best estimate methods, assumptions and data wherever possible so that the main results and insights are not unduly distorted by conservative assumptions.

Severe accident phenomena

A level-2 PSA requires the analysis of the complex interaction of physical and chemical processes that can occur in the course of the accident following core damage. This is based on an understanding achieved through an extensive number of experimental programmes conducted over the last two decades driven primarily by the TMI-2 accident. These experimental programmes were supplemented by intense computer code development and validation activities. The understanding is further augmented by information derived from a joint international cooperative project carried out under the auspices of OECD/NEA (commonly referred to as the Vessel Investigation Project) to investigate the end state of the TMI-2 reactor pressure vessel and the final lower head debris bed configuration.

An overview of key severe accident phenomena is provided in [1]. These phenomena are normally considered in two broad categories related to: (i) the accident progression and containment performance analysis, and (ii) the source term analysis (also referred to as the radiological release analysis).

The accident progression analysis serves two primary purposes: (i) to determine the event chronology and (ii) to identify and evaluate the challenges to the engineered barriers that have been incorporated to prevent a fission product release to the environment. These phenomena include: hydrogen generation and combustion; core-material relocation during the in-vessel phase of the severe accident; challenges to the reactor pressure vessel from molten core material; and challenges to the containment that can arise during the ex-vessel phase including overpressurisation of the containment and erosion of the containment basemat in light water reactor plants.

The source term analysis addresses the phenomena associated with the chemical processes affecting the radionuclide release and formation during the accident progression, and the transport of the radioactive material from the fuel through the containment to the environment. This analysis requires an in-depth understanding of the chemical and physical forms of the radionuclide species.

Following the TMI-2 accident, intense international research on severe accident phenomena took place with particular emphasis on the understanding of early containment failure mechanisms, and core melt and source term behaviour associated with this early phase. Many experimental programmes were established to improve the understanding of the phenomena that could occur and to provide data for both code development and validation – for example, the US NRC sponsored research programmes and the industry degraded core rulemaking (IDCOR) programmes sponsored by the US industry. Following the NUREG-1150 study, significant research programmes were also initiated in Europe (notably the PHEBUS programme), and continued largely through the co-operative research within the EC sponsored Framework Programmes. More recently, activities have focused on providing a better understanding of the phenomena associated with severe accident management measures and reducing the level of uncertainty for other phenomenological issues. An example is provided by the large scale melt coolability experiments performed in the MACE programme to get a better understanding of achieving ex-vessel melt coolability by water addition.

It is generally acknowledged that significant progress in the understanding of severe accident phenomena has been achieved over the last two decades and this is reflected by the gradual reduction of international research activities. This level of knowledge is deemed sufficient to enable the resolution of a number of key severe accident issues including: early containment failure caused by the missiles generated by an in-vessel steam explosion (generally referred to as α-mode containment failure); direct containment heating for most types of pressurised water reactors; and liner melt-through for some types of boiling water reactors. A thematic network project – EURSAFE – was initiated within the EC Framework Programme to achieve expert consensus on severe accident issues and to propose a structure to address any remaining key uncertainty issues. This research effort is continued in the severe accident research network of excellence (SARNET) underway within the present EC 6th Framework Programme. Internationally, some recent and ongoing collaborative research projects have been instigated under the auspices of the OECD to further improve the understanding in some key issues. These include: (i) the RASPLAV and MASCA programme on the large scale experimentation related to in-vessel molten corium pool behaviour, (ii) the MCCI programme on core concrete interaction and ex-vessel debris coolability, and (iii) the SERENA programme on fuel coolant interactions. The experimental data are also used for further validation and improvement of phenomenological models. Past OECD initiatives on the state of art reviews of some severe accident phenomenological issues can be found in [1]. In addition, ongoing OECD programmes can be found on the OECD/NEA web page (http://www.nea.fr/html/nsd/welcome.html).

It is also worth noting that level-2 PSA can provide useful feedback to the severe accident research community. For example, the insights of level-2 PSA have, in part, led to the pursuit (and subsequent completion) of significant hydrogen combustion research worldwide to address the issue of early containment failure as a result of hydrogen combustion phenomena.

Level-2 PSA methodology

As can be seen from [1], there is general agreement on the overall approach for carrying out a level-2 PSA. This typically involves the following steps:

- Development and quantification of the plant damage states (pdss) that form the interface between the level-1 and level-2 PSA.
- Accident progression modelling using a containment event tree approach.
- Containment performance analysis.
- Quantification of the containment event tree and categorisation of the endpoints into release categories.
- Radiological source term analysis for these release categories.
- Uncertainty analysis and sensitivity studies.

However, there are differences in the details of the analyses that have been carried out in relation to: the scope of the level-1 PSA that is used as the starting point for the level-2 PSA (for example, whether it includes external hazards and addresses low power and shutdown states); the number of the PDSs defined; the number of nodes and endpoints defined in the containment event trees; the number of source terms/ release categories defined; and the intended applications of the level-2 PSA. In principle, such differences will not impair the quality of the results as long as the relevant factors that influence the evolution of the accident are treated in the detail necessary for the individual steps and the information required in the subsequent steps is properly propagated.

It must be emphasised that the performance of deterministic severe accident analysis is an integral and significant part of a level-2 PSA and it is required to support the level-2 PSA process outlined above. Typically, preliminary severe accident analysis is also performed to allow meaningful definition and formulation of the steps listed above and, in particular, the source term categorisation.

It is also worth noting that alternative approaches for using severe accident analysis models in level-2 PSA are being explored. Such approaches, conceptually involving the coupling of a probabilistic driver to a deterministic accident progression analysis code, provide a natural, dynamic framework for addressing key phenomena (including actions carried out by the plant operators) and their uncertainties, and would likely affect the way expert judgment is brought into the analysis process. In order to facilitate the broad use of such approaches in practical applications, it appears that a number of issues will need to be addressed. Presuming the availability of a high-quality plant-specific model, these issues include the significant computational requirements associated with multiple runs of current deterministic analysis codes, the need for tools to dynamically focus analysis resources on risk-significant scenarios (for example, through intelligent, adaptive sampling), the potential need for new interfaces to couple the alternative level-2 PSA with the existing level-1 PSA, and the potential need to address credible phenomena not included in the deterministic analysis codes being used.

Definition of the plant damage states

The ranges of fault sequences that lead to core damage are identified in the Level 1 PSA. These need to be taken forward into the level-2 PSA which models how these fault sequences progress. Since there are a very large number of such fault sequences, they need to be grouped to make the subsequent accident progression analysis manageable. These groups, referred to as plant damage states (PDSs), are defined in terms of the attributes that would influence the way that the accident progresses to challenge the containment integrity and the release of radioactive material to the environment. The PDS attributes identified for pressurised water reactors typically include:

- The type of initiating event that has occurred (whether it is an intact circuit fault or a loss of coolant accident).
- The primary system pressure at the onset of core damage.
- The status of the safety systems (such as the emergency core cooling system) and support systems (such as electrical power and cooling water systems) as appropriate at the onset and as core damage progresses.
- The status of the containment protection and mitigation systems (such as the status of the containment itself; containment cooling and spray systems; hydrogen mixing/recombiners/igniters; containment venting).
- The integrity of the containment (that is, whether the containment is intact; containment isolation has failed; the containment has been

bypassed due to a steam generator tube rupture or an interfacing system LOCA that discharges outside the containment; there is significant leakage from the containment; or containment failure has already occurred at the onset of core damage leading to gross leakage).

The PDSs form the interface between the level-1 PSA and the level-2 PSA and define the initial and boundary conditions for the progression of the severe accident. In carrying out this grouping process, fault sequences categorised as core damage in the level-1 PSA are often identified as not leading to core damage (which can arise due to simplifying assumptions made in the level-1 PSA) and they need to be screened out.

The number of PDSs included in the analysis is usually in the range from 10 to 50 and they are defined using up to about 20 attributes. Examples of PDSs, the attributes used to define them and the grouping process used to reduce then to a practical number for analysis in the level-2 PSA can be found in [1]. A review of some recent European level-2 PSAs in the EC SARNET project showed a consistent approach has been applied in doing this. In a limited number of PSAs, a much larger number of PDSs (>100) has been defined based on a larger set of attributes and a finer grouping process.

The selection of the initial number of PDSs and the grouping into a final smaller, manageable number to be analysed in the level-2 PSA is based on considerations of the similarity in the anticipated severe accident progression behaviour. Typically, each PDS differs such that a containment event tree is quantified for each of them.

The current trend is to extend the scope of the level-2 PSA to include faults during the low power and shutdown phases of plant operation. This necessitates the identification of other attributes and PDSs to take account of the decay heat level, whether the reactor pressure vessel is closed or open, whether the primary containment is closed or open, and whether the irradiated fuel is in the reactor pressure vessel or the refuelling pool.

Accident progression analysis

This part of the level-2 PSA models the progression of the accident from core damage to the challenges to the containment and the subsequent release of radioactive material for each of the PDSs. This is generally carried out by using an event tree approach – referred to as either containment event trees (CETs) or accident progression event trees (APETs). These event trees need to model all the significant physical and chemical processes that could occur following a severe accident that challenge the containment or influence the release of radioactive material.

The nodes in the CET follow the chronology of the accident progression from core damage through failure of the reactor pressure vessel (RPV) to failure of the containment in the short term or in the longer term. The time frames, which are defined to mark the important stages of the severe accident progression and the times of major changes in the fission product release behaviour, typically include the following:

- From the occurrence of the initiating event up to the start of core damage.
- From the start of core damage but before failure of the RPV.
- Immediately following failure of the RPV.
- In the longer term when there is molten core material outside the RPV.

The containment event tree nodes are usually a set of questions that relate to whether particular phenomena occur in each of the time frames addressed in the analysis, whether any systems credited in the level-1 PSA have been recovered, whether severe accident management actions have been carried out and whether failure or bypass of the containment has occurred. Hence, an adequate number of time frames and nodes need to be defined to allow all the significant phenomena that are relevant during each time frame to be addressed.

The usual approach is to define a generic event tree structure that has the same time frames and asks the same nodal questions for each of the PDSs. However, the actual event trees that are drawn for individual PDSs will be different due to the different initial and boundary conditions defined by the PDSs. The endpoints of the event trees define the sequence of event that have occurred and the final state of the containment integrity. The grouping of the endpoints is discussed further in the section on source term analysis. The practice of using PDS specific CET structures is less common. However, a mixed approach has also been used where a small number of PDS-specific containment event trees have been defined that compliment the generic containment event trees. This can be convenient for modelling the response to a small number of PDSs for bypass sequences where most of the generic structure is not applicable.

Two approaches to constructing the event trees have evolved as follows: a small event tree approach where about 10 to 30 nodes are defined or a large event tree approach (as used in the NUREG-1150 study) where more than 100 nodes are usually defined. In addition, a common approach in the quantification of small event trees is to use a decomposition event tree (DET) or a phenomenological fault tree (PFT) to allow a more traceable and detailed treatment of the top event. The information contained in the two types of containment event tree is generally consistent because many of the event nodes of the large accident progression event trees are treated in the decomposition event trees used to support the smaller containment event trees.

The quantification of the event trees needs to be supported by information derived from several sources, including severe accident analysis, containment performance analysis, and fission product release and transport analysis. The assessment of the degree of uncertainty associated with the complex phenomena typically involves (in recent studies) the use of technical expert opinion either formally or in a less formal way.

Severe accident modelling

The physical and chemical processes that are expected to occur during severe accidents that govern its progression are complex and typically involve many simultaneous phenomenological interactions for which detailed experimental information may be sparse or not available. Hence mathematical modelling and computer simulation of these phenomenological processes needs to be carried out and this is influenced by the varying degrees of uncertainty.

There are generally three approaches adopted in the computer codes used for severe accident analysis:

- Stand-alone separate phenomena codes (also known as separate effects codes) which provide more detailed models of specific aspects of individual phenomena or a phase of the severe accident. For example, detailed modelling of steam explosions would require mechanistic treatment of the melt jet break-up during the pre-mixing phase, triggering, the propagation phase and the expansion phase giving the final thermal detonation. The aim is to develop analysis codes for nuclear power plants that are consistent with the state of the art and available experimental data.
- Integrated codes which address a set of key phenomena that occur during each specific severe accident sequence. They incorporate the thermal-hydraulic, chemical and fission product models into a single code for the core, primary and secondary coolant systems, and the containment building. These codes are designed to run relatively quickly so that they can process the large number of calculations necessary for the different severe accident sequences that arise for the different PDSs. To achieve this, they contain much simpler models than the separate phenomena codes. However, they often require expert judgement to assess the results of these codes.

- Simple parametric codes which use the results obtained mainly from the integrated codes expressed in terms of a number of parameters and allows for interpolation.

Since the NUREG-1150 study, significant progress has been made in the development of integrated severe accident analysis codes to model the complex melt progression behaviour and the resultant fission product release and transport behaviour. Two specific codes are widely used in the current generation of level-2 PSAs – MAAP (modular accident analysis program) and MELCOR. Both codes have undergone significant validation (based on both integral and separate effect experiments) and benchmarking exercises. Current application has tended to make use of these integral codes for providing the baseline analysis of accident sequences with supplementary analysis provided by the other standalone codes or expert judgement for the detailed evaluation of some phenomena.

Overall, the integral codes MAAP and MELCOR are seen to have reached a level of adequacy in providing a severe accident analysis capability for the understanding of overall plant behaviour and prediction of potential radiological releases to the environment. This is reflected by the scaling down of model development effort since the late 90s. Further refinements are confined largely to addressing issues related to the modelling of severe accident management measures and improving code performance (through user feedback). Users and experts acknowledge that a significant level of uncertainty related to some phenomena still exist (epistemic uncertainty) and this is generally addressed in the uncertainty analysis that is often carried out as part of the level-2 PSA.

Severe accident codes are also being developed in several countries including the thermal hydraulic analysis of loss-of-coolant, emergency core cooling and severe core damage code (THALES) which has been developed in Japan and the accident source term evaluation code (ASTEC) which is being jointly developed by IRSN of France and GRS of Germany and can be seen as a European code for future severe accident analysis.

Experimental and analytic studies (including benchmark exercises) are being conducted at the international level in order to improve the confidence in the predictive capabilities of the models embedded in the various severe accident computer codes.

Containment performance analysis

There are a number of ways in which the containment integrity could fail. Two of them, containment isolation failure and containment bypass, are usually modelled in the level-1 PSA since they relate to the containment status at the onset of core damage. These failure modes are included in the definition of the PDSs. For those PDSs in which the containment is intact, the level-2 PSA addresses how the containment would behave due to the loading placed on it as a result of the physical and chemical processes that occur following core damage, and determine whether it would fail. The mechanisms challenging the containment function and resultant failure modes are listed in [1]. They include:

- Rapid overpressurisation (due to a steam explosion, hydrogen combustion or direct containment heating).
- Slow overpressurisation (due to the continuous generation of non-condensable gases and steam).
- High temperatures in the containment in the longer term.
- Containment bypass due to failure of steam generator tubes caused by creep rupture.
- Missile impact (following energetic events inside the containment).
- Containment under-pressure caused by temporary depletion of non-condensable gases in the containment atmosphere (for example, due to venting, an unisolated leak or hydrogen combustion) followed by operation of the containment cooling systems.
- Erosion of the containment basemat and the liner (due to contact with molten core material).

In the early level-2 PSAs, a "threshold" model was adopted to characterise the loss of containment integrity. This involved the definition of a threshold pressure, with some associated uncertainty range, above which gross failure of the containment was assumed to occur. The containment shell was assumed to fail catastrophically within a narrow band of pressures.

More recently, detailed containment integrity studies have been carried out using finite element modelling techniques supported by numerous experiments based on scale models of containment structures that were pressurised to failure. These have shown that the leak-before-break scenario is more likely and local leakage failure is the more likely failure mode. The dominant failure mechanism depends on the details of the design of the containment. For example, for containments equipped with liners, the failure mechanism could be (i) a liner rupture caused by the interaction of the liner and its anchorage system with the concrete at major stiffness discontinuities or (ii) failure at the containment access hatch. Work has also been carried out by NEA/CSNI on an international standard problem on containment integrity (ISP48) which provided further international effort to extend the understanding of the response of actual containment structures to pressure loading and to compare analytical predictions to measured behaviour. This ISP, based on a 1:4 scale model of a pre-stressed concrete containment vessel constructed and tested at Sandia National Laboratories, involved the prediction of the structural response of the model to static and transient pressure and thermal loading.

In order to provide a more realistic assessment of containment performance and pathways for the radionuclide release to the environment, more recent level-2 PSAs have included a plant-specific containment performance analysis. This is based on a structural analysis to determine how the containment will behave due to the pressure/temperature conditions that could arise and criteria that relate to when failure will occur. The analysis relates to the actual design of the containment and takes account of potential leakage paths provided by the doors, penetrations, seals and other possible weak areas. The analysis identifies the different failure modes and the corresponding failure sizes. The results are in the form of fragility curves as a function of pressure and temperature. At the end of each time frame, the status of the containment is assessed and can be represented by (i) the containment is intact with the normal design basis leakage, (ii) there is enhanced leakage or (iii) gross failure has occurred.

Quantification of the level-2 PSA model

The next stage of the level-2 PSA is to quantify the analysis to determine the frequency of the various fault sequences identified in the containment event trees. The data required for this is the frequencies of the PDSs, which are derived in the level-1 PSA, and the conditional probabilities of the event tree branch points.

There are differences in the meaning of the nodes of the containment event tree and the way that they are quantified. These include the following: failure of safety systems such as the containment spray system that are quantified using fault trees; structural failures of the containment that are quantified using a model of the performance of the structure; and the occurrence of physical phenomena where the split fractions relate to the analyst’s “degree of belief” that a particular phenomenon will occur.

Where the assigned probabilities represent the analyst’s “degree of belief” that an event X will occur in timeframe Y given the set of accident conditions, the numerical values are derived from judgement which is supported by available sources of information. An attempt to make this judgement process more traceable is achieved by making use of supporting analysis such as such as decomposition event trees and phenomenological fault trees where possible. Although this aspect of the quantification appears subjective, it is firmly established as part of the overall PSA methodology.

The quantification of the event trees also needs to take account of the interdependencies between the nodes in the event trees. These can arise due to dependencies between the support systems, the phenomena that could occur in successive time frames and between human actions in carrying out the severe accident management actions.

Recent notable developments to overcome the current limitations include the risk oriented accident analysis methodology (ROAAM) and the use of physical models in the French IRSN level-2 PSA methodology [2]. Attempts are also being made to relate the numerical scaling for the event tree branch probabilities (that is, the so called split fractions) to the state of the knowledge (that is, the epistemic uncertainty). Unlike the level-1 PSA, there are no data for phenomena that have not occurred and thus no recognised and authoritative level-2 PSA databases for underpinning the quality of judgement. Many recent PSAs still rely on the arguments given in NUREG-1150 since this is seen as a reference study. At present, there is no known concerted action to produce such a database based on consensus of the current understanding of the phenomena although some initial initiatives were instigated within the EC Framework Programmes. This included reviews of experimental and analytical data gathered for some phenomenological issues but did not extend to relate to split fractions used in level-2 PSAs. However, it must be recognised that a significantly improved understanding of certain phenomena has been achieved since the NUREG-1150 study was carried out and this should be reflected in the current performance of level-2 PSAs.

The development and quantification of the CETs require that a large number of plant and containment states are handled and there are a number of computer codes available that can do this. These include:

- Codes developed specially for the level-2 PSA (such as EVNTRE which was developed for the NUREG-1150 study, and KANT which has been developed by IPSN in France).
- Level-1 PSA codes that handle linked event trees and fault trees that can also be used for the level-2 PSA (such as risk spectrum). One of the advantages of doing this is that the same software can be used for an integrated level-1 and Level-2 PSA.

The quantitative results of the level-2 PSA are the frequencies of the release categories defined in the analysis (and the uncertainties in these frequencies where an uncertainty analysis has been carried out). However, the most common result presented is the large release frequency (LRF) or the large early release frequency (LERF). In this context, "large" is defined as being greater than a specific quantity of radioactive material which is often defined in terms of a fraction of the radioactive inventory of the core and "early" relates to the release occurring before the effective implementation of the off-site emergency response and protective actions so that there is the potential for early health effects. These can be compared with the probabilistic targets for LRF and LERF (where they have been defined). However, there is no consensus in the member countries on what constitutes a large/early release.

Source term analysis

The large number of CET end-points will need to be grouped to provide the interface between the level-2 PSA and the level-3 PSA consequence analysis. The categorisation scheme is usually comprised of two distinct steps. The first groups the CET end-points on the basis of similar source term phenomena to form source term categories (STCs) and the second groups STCs on the basis of similar environmental consequence to form release categories (RCs). The allocation of STCs to RCs is based on the potential of each source term to cause adverse effects. As relatively few of the current level-2 PSAs are extended to level 3, the term RCs is typically used for the direct grouping of CET end-points. It should be noted that as the industry moves towards new reactor designs with different release characteristics, more explicit level-3 analyses may be required.

The CET end points are categorised according to a number of attributes related to fission product release, retention and transport mechanisms through each of the major barriers to the environment. The purpose of this categorisation (also referred to as source term binning) is to allow practical source term analysis to be performed for each defined RC. The key attributes include:

- The timing of the release.
- The status of the containment (that is, whether containment isolation has occurred, whether containment failure has occurred giving rise to enhanced leakage or a large leakage area, and whether molten core material released from the reactor pressure vessel is challenging the integrity of the basemat).
- The way that the release is occurring (such as high pressure melt ejection, dry core concrete interaction, and core concrete interactions from submerged corium).

- The fission product removal mechanisms (such as containment sprays, or retention in the secondary containment or reactor building).
- The pressure suppression pool (for boiling water reactors).

If the level-2 PSA is to be taken into a level-3 PSA, additional attributes may need to be defined and they include the height of the release, the energy of the release and the release duration.

A source term – that is, the quantity and duration of radionuclide release, is assigned to each STC. The source terms used in the NUREG-1150 study were based on the parametric XSOR code developed for the study. With the advances made in the development of integrated severe accident codes, the source terms for specific sequences are generated directly in recent level-2 PSAs. In these codes, the major radionuclide species are grouped on the basis of similarity in chemical and physical properties and these default groupings are similarly adopted in the PSAs. These source terms may be adjusted to account for key aspects not explicitly modelled in the codes. Typical examples are for the treatment of the source terms for energetic events and the consideration of the formation of organic Iodine, which may become important for the management of the later phases of a severe accident.

In some PSAs, the source term estimates are confined to the noble gases, Iodine and Caesium groups as I/Cs releases provide an indication of the early and latent human health consequences. Noting that other elements (and other chemical forms of the elements) can impact the offsite consequences, the need for detailed accounting of fission product species is dependent on the objectives and scope of the level-2 PSA. A summary of the derivation of RCs and the attendant source terms for a number of European level-2 PSAs is provided in [8].

Expert judgement

In the level-2 PSA, there are many areas where it is not possible to carry out a definitive analysis or there exists considerable phenomenological uncertainty due to a lack of understanding and knowledge. This arises where there is no accepted state of the art, or the relevant data do not exist or exhibit a high variability. In these cases, the analysis normally relies on some degree of expert judgement or expert consultation.

As part of the NUREG-1150 methodology, formal elicitation of expert judgement on the issues for which the uncertainty was seen as greatest was adopted. Six expert panels were established for the formal elicitation process (dealing with the front end, in-vessel, containment loads, structural response, molten core-concrete interactions and source term). The framework is outlined in [1]. The approach included elicitation training for the experts, the elicitation process and the aggregation of results.

The formal NUREG-1150 elicitation approach, which is intended to control potentially important sources of error and bias, can require significant resources to carry it out fully. Recent level-2 PSAs have generally used less formal methods. Recognising that the need for a formal analysis can be analysis-dependent, NUREG/CR-6372 describes two elicitation approaches involving varying degrees of formality. (Although this was written to support seismic hazard analysis, much of the material in that report is sufficiently general to be applicable to a wide range of problems requiring elicitation).

An attempt was made recently in an EC Framework Programme project to carry out a benchmark exercise on expert judgement in level-2 PSA (BEEJT) by applying five structured expert judgement methodologies to different problems. The first benchmark exercise involved the blind prediction for a melt quenching experiment performed at the FARO facility and the second involved the assessment of hydrogen combustion based on a loss of offsite power sequence for a reference PWR. The study was very much focussed on the assessment of the quality characteristics of the expert judgement techniques against some criteria (for example, the applicability and traceability of methods) in the two benchmark activities.

Uncertainties

The uncertainties inherent in the level-2 PSA arise in two ways as follows:

- Aleatory uncertainties that arise from the natural randomness in the processes that occur during a severe accident. This type of uncertainty cannot be reduced or eliminated.
- Epistemic uncertainties that arise from the lack of knowledge in the processes that occur during a severe accident. This type of uncertainty can be reduced or eliminated by gaining a better understanding from further research and development.

In the context of a level-2 PSA, the epistemic uncertainties can be classified into three types as follows:

- Parameter uncertainty in the probabilities used to quantify the containment event trees.
- Model uncertainty due to the incomplete knowledge of the phenomena that can occur during a severe accident, or to inadequacies or simplifications in the modelling.

- Completeness uncertainty due to the incompleteness of the analysis - that is, whether there are any fault sequences and specific phenomena associated with these sequences that have not been included. This type of uncertainty can be reduced by carrying out a peer review of the analysis.

To some extent, the level-2 PSA addresses uncertainties directly since, in the quantification of the analysis, the probabilities of the branch points of the event trees relate to the analyst's degree of belief in the possible outcomes given the uncertainties involved. The containment event tree methodology of the level-2 PSA thus can be regarded as a suitable framework for addressing parameter and model uncertainties explicitly. More recently, the ROAAM approach may be seen as providing a more detailed structured approach in dealing with epistemic uncertainties associated with a number of severe accident issues and has been used in a complementary way to level-2 PSAs. Advanced methodologies in the form of dynamic event tree methods are also being developed to deal with the uncertainty presented by stochastic events [3].

The NUREG-1150 study provided a milestone in level-2 PSA methodology in that it included a structured uncertainty analysis as outlined in [1]. However, this approach has only been applied in a limited number of cases. In the majority of level-2 PSAs, the uncertainty analysis has been pursued largely in the form of simple sensitivity studies for the containment event tree analysis. Although this represents a practical approach to the consideration of uncertainties, the results of the sensitivity studies have no statistical significance in the overall level-2 PSA.

The process of quantification of uncertainties and assessment of their relative importance has always been regarded as an essential and integral element of the overall PSA methodology, although the manner and extent in which uncertainties are addressed in PSAs can differ considerably. The overall objective of the uncertainty analysis in the PSA is to provide a measure of the imprecision in the PSA outcomes (PDS frequencies, STC/RC frequencies or the ultimate risks), and the overall objective of the sensitivity analysis is to identify the major contributors to this imprecision. Although the tools and methods are available to carry out an uncertainty analysis, the treatment of uncertainties in the context of an entire PSA is quite a resource intensive exercise that cannot be achieved in a straightforward manner. In deciding the method of uncertainty analysis for a level-2 PSA, the nature of the uncertainties that need to be addressed in the CET analysis, severe accident progression analysis and source term analysis must be considered. Depending on the requirement for the uncertainty analysis in the overall PSA, the choice of method will also depend on the need to achieve compatibility with the other components of the PSA.

It is clear that progress has been made recently in methodology development as attempts are made to address uncertainties in level-1 and level-2 PSAs in a systematic and integrated manner. This is reflected by the advanced techniques presented in the proceedings of [3] and [6]. The potential of these methods is being investigated in some PSAs (for example, the response surface approach in the IRSN level-2 PSA and the KAERI methodology [6] in the formal integration of level-1-2 PSA uncertainties). However, it needs to be recognised that the way that uncertainties will be addressed any specific PSA will ultimately depend on its objectives and scope.

Use of the level-2 PSA for severe accident management

The results of the level-2 PSA can also be used to identify the principal contributors to the risk and changes that can be made to the design or operation to reduce the risk. This decision making process needs to take account of the significant phenomenological uncertainties inherent in the level-2 PSA.

Severe accident management measures include the provision of hardware and the development of procedures or structured guidance that identifies the actions to be carried out in order to return the plant to a controllable state and mitigate the consequences of an accident. These measures include:

- Preventative accident management measures which are carried out during the evolution of an event sequence before the design basis is exceeded. Their aim is to prevent core damage and containment bypass sequences.
- Mitigative accident management measures which are carried out after core damage has occurred. Their aims are (i) to prevent the accident leading to failure of the reactor pressure vessel or the containment, and (ii) to control the transport and release of radioactivity material with the aim of minimising off site consequences.

The hardware that has been incorporated to mitigate the consequences of severe accidents for nuclear power plants includes:

- Hydrogen control systems that have the capacity to cope with the rate of hydrogen generation after core damage.
- Filtered containment venting systems to prevent overpressurisation of the containment in the longer term.
- Dedicated systems for retaining and cooling molten core material outside the reactor pressure vessel.

Examples of actions that could be carried out include:

- Opening the pressuriser relief valves to reduce the primary circuit pressure and avoid molten core material being ejected from the reactor pressure vessel under high pressure.
- Adding water to the containment by any available means after the molten core has exited from the primary circuit to provide a cooling mechanism.

Insights from level-2 PSAs, including an understanding of specific plant vulnerabilities to severe accidents, have enabled the development of severe accident mitigation procedures or structured guidance – referred to as severe accident management guidance (SAMG). Many nuclear power plants worldwide are now equipped with this capability driven partly by the development of the generic SAMGs by vendor owners groups. This severe accident management provision forms part of the plant emergency management system which need to provide clearly defined interfaces and responsibilities for decision making in the unlikely event of a severe accident. The mitigation measures to be adopted need to be compatible with the equipment, instrumentation and diagnostic aids that are available to the plant operators and technical support staff.

The use of level-2 PSA models to support severe accident management development is illustrated further in the assessment of in-vessel retention strategy for boiling water reactors and the Korean APR1400, and the hydrogen management strategy for the Loviisa plant in Finland [3, 6].

Discussion

Many level-2 PSAs have been performed in recent years and they are now seen as an integral part of nuclear power plant safety cases. A consistent framework has been developed for carrying out the analysis and this is made up of the steps described earlier. However, there are differences in the way that these steps have been carried out and in the level of detail of the analysis, which is partly attributable to the need to fulfil the objectives set for particular level-2 PSAs. Some of these differences have been identified through an in-depth comparison of the modelling approaches adopted in a number of Level-2 PSAs undertaken in Europe – see [7]. The reconciliation of some of these differences cannot be achieved in a straightforward way and some of these issues are likely to be addressed in the ongoing development of the American Nuclear Society standard on level-2 PSA. The following two areas, pertinent to the current generation of reactor designs, can be seen as key future activities placing additional demands on further model development:

- Integration of the level-2 PSA for living PSA and risk-informed applications. This integration is to provide a plant tool to support plant operation and prioritisation/justification of plant improvement, including the increasing demands of power uprating and utilisation of higher burn-up fuels. Such a plant tool will require further development in the level-2 PSA modelling in some areas including improved and consistent treatment of the level-1/level-2 PSA interface, safety system recovery and human reliability analysis (HRA). An example of such development is the living level-2 PSA for the Finnish Olkiluoto boiling water reactors [3]. Note that, in most current level-2 PSA studies, HRA plays a far lesser role than it does in level-1 PSA. Recognising the general importance of human contributions to accidents and accident risk, and the desire to assess the effectiveness of severe accident management measures, the development and application of HRA methods for level-2 PSA appears to be a potentially important future activity.

- Formal treatment of level-2 uncertainties, including integration of level-1/level-2 uncertainties. The quality of a PSA to support decision making is underpinned by a systematic evaluation of the impact of key issues of uncertainty on the results. Models of varying degree of formalism and sophistication have been developed and applied to a number of level-2 PSAs (see [6]), ranging from simple sensitivity studies to a more detailed treatment of epistemic and aleatory uncertainties in a structured approach involving the propagation of these uncertainties. The development of implementation guidelines in this area in meeting the specific demands of a level-2 PSA is seen as a current priority.

Conclusions

The main message of this Technical Opinion Paper is that the level-2 PSA methodology may now be seen as mature. This is reflected by the large number of high quality analyses that have been performed in recent years and used to identify the potential vulnerabilities to severe accidents and the accident management measures that could be implemented.

The level-2 PSA is now seen as an essential part of the safety analysis that is carried out for all types of nuclear power plants worldwide. The information provided by the level-2 PSA is being used by plant operators and Regulatory

Authorities as part of a risk informed decision making process on plant operation and more specifically on issues related to severe accident management.

A consistent framework has been established with the development of the individual components of the level-2 PSA methodology and guidance has been produced by international organisations for carrying out the analysis. In practice, however, there are still differences in the approach and the level of detail in the individual steps that have been carried out in different analyses, partly due to the different objectives that have been defined for these studies. Quality standards and guidelines are currently being developed for level-2 PSA which should address many of these differences.

The acceptability of the methodology since the early studies in the 1980s is due largely to the significant progress made in the understanding of severe accident and source term phenomenology and in the model development in the current generation of integrated severe accident analysis codes. The research and development activities have continued internationally, albeit at a reduced scale, with emphasis on improving the state of knowledge and providing further data for model validation and improvement.

Further development in level-2 PSA is likely to see its integration within a Living PSA and its use for risk-informed applications. This requires improvement in the level-2 PSA methodology in a number of areas, including: the level-1/level-2 PSA interface, the modelling of safety system recovery and human reliability analysis.

The epistemic uncertainty related to some level-2 PSA issues is regarded as being quite large. The impact of this on risk-informed decision making will also require further consideration of uncertainty treatment in a more integrated manner.

Finally, given the role that integrated severe accident codes (supported by research) have played in the acceptance of level-2 PSA, future level-2 PSA research and development activities should be aimed at making these codes play a more central and integral role in the PSA quantification process. Such a shift is likely to alter (and quite possibly diminish) the role of expert judgement and phenomenological event tree modelling in the quantification.

References

[1] Level 2 PSA Methodology and Severe Accident Management; NEA/CSNI/R(97)11; NEA/ CSNI (ed.); 1997.

[2] *Severe Accident Risks: An Assessment of Five US Nuclear Power Plants*; NUREG 1150; USNRC; 1990.

[3] *OECD/CSNI International Workshop on Level 2 PSA and Severe Accident Management*; Cologne, Germany; 29-31 March 2004.

[4] *Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2)*; IAEA Safety Series No 50-P-8; IAEA; 1995.

[5] *Regulatory Review of Probabilistic Safety Assessment (PSA) Level 2*; IAEA-TECDOC-1229; IAEA; July 2001.

[6] *OECD/CSNI Workshop on Evaluation of Uncertainties in Relation to Severe Accidents and Level 2 PSA*; Aix-en-Provence, France; 7-9 November 2005.

[7] *Towards Harmonisation of Level 2 Probabilistic Safety Assessment in Europe*; Post FISA 2006 Workshop; Luxembourg; 16 March 2006.

[8] *Overview of SARNET Progress on PSA 2 Topic, The First European Review Meeting on Severe Accident Research (ERMSAR)*; B Chaumont; Aix-en-Provence; 14-16 November 2005.

OECD PUBLICATIONS, 2 rue André-Pascal, 75775 PARIS CEDEX 16
Printed in France.