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FOREWORD

Probabilistic safety assessment (PSA) is increasingly being used as part of the decision making process to assess the level of safety of nuclear power plants. The methodologies in use are maturing and the insights gained from the PSAs are being used along with those from deterministic analysis.

Many regulatory authorities consider the current state of the art in PSA to be sufficiently well developed for results to be used centrally in the regulatory decision making process — referred to as risk informed regulation.

For these applications to be successful, it will be necessary for the regulatory authority to have a high degree of confidence in the PSA. However, at the 1994 IAEA Technical Committee Meeting on Use of PSA in the Regulatory Process and at the OECD Nuclear Energy Agency Committee for Nuclear Regulatory Activities (CNRA) "Special Issues" meeting in 1997 on Review Procedures and Criteria for Different Regulatory Applications of PSA, it was recognized that formal regulatory review guidance for PSA did not exist. The senior regulators noted that there was a need to produce some international guidance for reviewing PSAs to establish an agreed basis for assessing whether important technological and methodological issues in PSAs are treated adequately and to verify that conclusions reached are appropriate.

In 1997, the IAEA and OECD Nuclear Energy Agency agreed to produce, in co-operation, guidance on Regulatory Review of PSA. This led to the publication of IAEA-TECDOC-1135 on the Regulatory Review of Probabilistic Safety Assessment (PSA) Level 1, which gives advice for the review of Level 1 PSA for initiating events occurring at power plants. This TECDOC extends the coverage to address the regulatory review of Level 2 PSA.

These publications are intended to provide guidance to regulatory authorities on how to review the PSA for a nuclear power plant to gain confidence that it has been carried out to an acceptable level of quality so that it can be used as the basis for risk informed decision making within a regulatory decision making process. They give advice on how to set about reviewing a PSA and on the technical issues that need to be addressed.

It is intended that further work will be carried out in the future to extend the coverage of the report to accident sequences occurring at low power and shutdown states, and for Level 3 PSA.

The IAEA appreciates the work performed by all the participating experts and wishes to thank them for their valuable contribution to the preparation of this report. The IAEA officers responsible for this publication were V. Ranguelova and A. Gómez Cobo of the Division of Nuclear Installation Safety.

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1. INTRODUCTION

1.1. BACKGROUND

Probabilistic safety assessment (PSA) of a nuclear power plant provides a comprehensive, structured approach to identifying failure scenarios and deriving numerical estimates of the risks to workers and members of the public. PSAs are normally performed at three levels as follows:

- **Level 1 PSA** which identifies the sequences of events that can lead to core damage, estimates the core damage frequency and provides insights into the strengths and weaknesses of the safety systems and procedures provided to prevent core damage.
- **Level 2 PSA** which identifies the ways in which radioactive releases from the plant can occur and estimates which identifies their magnitudes and frequency. This analysis provides additional insights into the relative importance of the accident prevention and mitigation measures such as the reactor containment.
- **Level 3 PSA** which estimates public health and other societal risks such as contamination of land or food.

The emerging standard in the last few years is for Level 2 PSAs to be carried out. A large number of such analyses have been carried out worldwide (in 1997, this included 19 for PWRs and 8 for BWRs described in the review of the state of the art carried out by OECD/CSNI — see Ref. [1]).

The PSA provides a systematic approach to determining whether the safety systems are adequate, the plant design is balanced, the defence in depth requirement has been realized and the risk is as low as reasonably achievable.

In particular, Level 2 PSAs have been carried out for the following reasons:

- to provide insights into how the plant would behave during a severe accident,
- to identify weaknesses in the level of protection provided for severe accidents,
- to identify additional safety systems and accident management measures that would provide further protection against severe accidents, and
- to provide an input into emergency preparedness.

The Level 2 PSA needs to address all the phenomena that could occur during and following core damage, which have the potential to challenge the integrity of the containment and lead to a significant release of radioactive material to the environment.

PSA is increasingly being used as part of the decision making process to assess the level of safety of nuclear power plants. The methodologies have matured over the past decade or so and, while they are continuing to develop, PSA is now seen as a very useful, and often essential tool to support the deterministic analysis which has traditionally been carried out. The insights gained from the PSA are being considered along with those from the deterministic analysis to make decisions about the safety of the plant. Additionally, many regulatory authorities consider the current state of the art in PSA to be sufficiently well developed for results to be used centrally in the regulatory decision making process — referred to as risk informed regulation.

For these applications to be successful, it will be necessary that the PSA provides the required support and for the regulatory authority (and the utility) to have a high degree of confidence in the PSA.

The use of PSA in the regulatory process was the subject of several IAEA consultants and technical committee meetings and two OECD Nuclear Energy Agency (NEA) Committee for Nuclear Regulatory Activities (CNRA) ‘Special Issues’ meetings — see Refs [2] and [3]. At these meetings, the senior regulators agreed that the use of PSA as a tool in the regulatory decision making process is increasing and it is now becoming acceptable to use PSA as a complement to the deterministic approaches to address plant safety concerns.

Although the current trend is for regulatory authorities to move towards a more risk informed approach to their activities, it was found that there is considerable variation in the way they carry out their assessments of PSAs. While many countries have already established, or are planning to establish, guidance for reviewing PSAs, it is often not a formalized or standard type of practice. Some international guidance is available but this is applicable to a specific purpose — for example, the International Peer Review Service (IPERS) guidelines [4] produced by IAEA as the basis for the service it provides to its Member States in the peer review of PSAs.

The senior regulators concluded that there was a need to produce some international guidance for reviewing PSAs. The main objective of this guidance would be to establish an agreed basis for assessing whether important technological and methodological issues in PSAs are treated adequately and to verify that conclusions reached are appropriate.

This co-operative effort led to the publication of IAEA-TECDOC-1135 on the Regulatory Review of Probabilistic Safety Assessment (PSA) Level 1 in February 2000. The present document follows on from this effort and provides regulatory review guidance for Level 2 PSAs.

1.2. REGULATORY REVIEW OF PSA

The PSAs that are currently produced provide unique insights into the way initiating events and safety systems interact and give an overall picture of plant behaviour. In particular the Level 2 PSA provides insights into containment responses to severe accidents. These insights are of value to both the plant operators and the regulatory authority.

This increasing use of PSA has led to the realization that the production and use of a PSA requires substantial efforts by both the utility and the regulatory authority to carry out and review them. In addition, there is a need to provide knowledge and training to personnel in the use of these methods.

Inherent in the production and review of a PSA is the ability of those involved to determine what is acceptable. The objective of a regulatory review is to provide confidence in the PSA study to ensure that it is fit for its intended purpose. As industry further develops the use of PSA in justifying plant changes and modifications, the regulatory authority and other agencies need to understand how the PSA has been produced in order to be able to assess its applicability in the decision making process. The review process becomes an extremely important phase in determining the acceptability since this provides a degree of assurance of the scope, validity and limitations of the PSA, as well as a better understanding of the plant itself. This is becoming increasingly important with the advent of a risk informed regulatory decision making environment. Additionally, utility involvement is important, since the prime responsibility for the safety of the plant rests with the utility and not with the regulatory authority. Therefore, motivation exists on the part of both the regulatory authority and the utility to ensure that PSAs are performed adequately.

In preparing this document, it is recognized that differences exist between countries in the way that the nuclear industry is organized — including the utilities, operators, designers and manufacturers. In this document, the term ‘utility’ is used and is taken to encompass the industry as a whole. In addition, there are differences in the way a regulatory authority operates in that, in some countries, it is completely within the governmental system while, in others, it is outside government but responsible to, or licensed by, it. These differences are reflected in the way that the PSAs are produced and reviewed in different countries.

The review of the PSA may be performed by the regulatory authority alone or with outside consultants or even in some cases with the help of international peer reviews. The guidance provided covers all these possibilities.

1.3. SCOPE OF THE REPORT

This publication provides recommendations on how to carry out the regulatory review of a Level 2 PSA produced by a utility. It is intended to be general and to be applicable to the review of a PSA for any power reactor type. By following the guidance given, the regulatory authority has to be able to satisfy itself that the Level 2 PSA has been carried out to an acceptable level of quality and that it can be used for its intended applications.

This publication needs to be read in conjunction with Ref. [5], which gives guidance for the review of Level 1 PSA for initiating events occurring during full power operation. The review process set out in Ref. [5] for Level 1 PSA also applies to Level 2 PSA and this is not repeated here; only differences and additional requirements are given. It also gives more detailed guidance on the Level 1/Level 2 PSA interface than was provided in Ref. [5]. It is intended that further work be carried out in the future to provide review guidance for event sequences occurring at low power and shutdown states, and for Level 3 PSA.

As a result of carrying out a Level 2 PSA, changes are often identified which would increase the level of safety. This might include the incorporation of specific safety systems that provide protection against the consequences of a core melt or other accident management measures. In reaching the decisions on what improvements will actually be made, the insights gained from the PSA are combined with those gained from the deterministic analysis and other factors (such as the cost, the remaining lifetime of the plant, etc.). The review of this decision making process is not within the scope of this publication.

1.4. STRUCTURE OF THE REPORT

This report gives key recommendations for carrying out the regulatory review of a Level 2 PSA.

Section 2 gives guidance on how the regulatory authority carries out the review of the Level 2 PSA and addresses issues such as:

- approach to the review,
- reviewing the purpose and scope of the Level 2 PSA,
- reviewing the methods and assumptions used, and
- auditing the Level 2 PSA production process.

The differences in the review process for Level 2 PSA from those described in Ref. [5] for Level 1 PSA are also highlighted in Section 2.

Section 3 gives guidance on the technical issues that need to be addressed in carrying out the review of a Level 2 PSA. This covers the main tasks as follows:

- familiarization with the design and operation of the plant and provision of the plant data which is important for severe accidents,
- interface with the Level 1 PSA and the grouping of the accident sequences leading to core melt into plant damage states for the Level 2 PSA,
- analysis of the progression of the severe accident taking account of the various phenomena that could occur,
- analysis of the performance of the containment following the loadings which arise as a consequence of the severe accident,
- developing a probabilistic modelling framework — usually an event tree analysis, which models the development of the accident sequences that could occur,
- quantification of the event sequences identified. This involves both deterministic analysis and expert judgement and to address the myriad uncertainties in the severe accident phenomena that could occur,
- characterization of the source terms for the radioactivity released to the environment. This requires modelling of the release of radionuclides from the molten fuel and their transport out of the containment, and
- interpretation and use of the results of the Level 2 PSA analysis.

Some of the issues addressed in Ref. [5] such as sensitivity studies/uncertainty analysis and the validation/verification of the computer codes used in the analysis are also particularly relevant to Level 2 PSA and hence they are also addressed here.

A list of references, which provide more detailed guidance on many of the Level 2 PSA issues, is provided at the end of the report. Abbreviations used in the report and the names of those who contributed to the production of this TECDOC are also given at the end of this publication.

In preparing this report, it has been recognized that there are differences in the terminology used in different countries and, whilst every attempt has been made to use consistent terminology throughout, readers should take these differences into account in applying the guidance given.

In the report, the term PSA, for probabilistic safety assessment, is used throughout. This is taken to be the same as PRA (probabilistic risk analysis/assessment) and the two are considered to be interchangeable. In addition, it is recognized that there are differences in the way that the industry is set up and that terms such as ‘utility’, ‘plant operator’ and ‘licensee’ may mean different things in different countries. In producing this regulatory guidance document, these terms are considered to be interchangeable and ‘utility’ is used throughout. In addition, there are differences in who actually carries out the PSA. In this document, the view is taken that the PSA is carried out by the ‘utility’, since it is the responsibility of the utility, although it is often carried out by the plant designers or sometimes by consultants.

2. THE REVIEW PROCESS

2.1. INTRODUCTION

This section gives recommendations on the way a regulatory authority should set about reviewing the Level 2 PSA for a nuclear power plant to gain confidence that it has been carried out to an acceptable level of quality.

In providing this information, it is recognized that the approach to the regulation of nuclear power plants in general and to the regulatory review of PSA may be different in different countries. In addition, the approach may also be different depending on the purpose of the review — for example, the review that is carried out on the PSA for a new reactor design may be different from that for an existing reactor carried out as part of a periodic safety review.

Guidance is given on:

- the approach to the review,
- the review of the aims, objectives and scope of the PSA,
- the review of the methods and assumptions used in the PSA, and
- the review/audit of the utility's PSA production process.

2.2. APPROACH TO THE REVIEW

The approach to the review of a Level 2 PSA is very similar to that for a Level 1 PSA and the reader needs to refer to Ref. [5] for general guidance. However, there are additional considerations that relate specifically to the review of a Level 2 PSA; these are set down below. (The section headings are the same as those in Ref. [5]).

2.2.1. Timing of the review

Ref. [5] considers the advantages and disadvantages of 'off-line' and 'on-line' reviews. As for the Level 1 PSA, it is considered that the most efficient way is for the regulatory authority to carry out an on-line review of the Level 2 PSA whenever possible so that specific tasks are reviewed as they are completed rather than wait for the whole of the analysis to be completed. This would allow the regulatory authority to determine whether the analysis is being carried out in an acceptable way and, if not, ensure that any deficiencies are rectified at an early stage. However, it is recognized that there are situations where 'off-line' review may be chosen for particular reasons.

Having agreed on the timing of the review, it is suggested that a schedule of work be drawn up with the utility's PSA team that fits the needs of both organizations, ensures that the review process is conducted efficiently and that any delays in completing the PSA or the review are minimized.

The review of the Level 2 PSA would normally follow on from the review of the Level 1 PSA. This is important since any deficiencies in the Level 1 PSA will be transmitted to the Level 2 PSA and may lead to incorrect conclusions.

2.2.2. Extent of the review

As for the Level 1 PSA (see Ref. [5]), the extent of the review of the Level 2 PSA will need to be decided at the start of the review process. This can range from an extensive review of the assumptions, models and data contained in the Level 2 PSA, to a much more limited review. The choice here depends on factors such as the intended use of the PSA.

Although the extent of the review will depend on national practices and other factors, it will need to be sufficient to provide the regulatory authority with the level of confidence it is seeking. In particular, it will need to provide confidence that the analysis is consistent with the current state of the art (as defined in Refs. [1] and [6]), and that it has addressed all the significant phenomena which would affect the accident progression and the magnitude of the source term released from the plant.

Other factors which might influence the scope of the review might include the level of risk from the plant, the experience with that reactor system, whether an Independent Peer Review of the analysis has been carried out and whether it is intended to use the Level 2 PSA as a basis for risk informed decision making. However, for many regulatory authorities, the review of the Level 2 PSA may be an excellent source of additional knowledge about the response of the plant to severe accidents that, by itself, may justify an extensive review in any case.

In all cases, the focus of the review is on the issues that are important to determine the response of the plant to severe accidents. Even in an extensive review, it is not necessary to independently verify every detail.

2.2.3. Documentation for the review

As for the Level 1 PSA (see Ref. [5]), the starting point for the review is the set of documentation, which describes the design and operation of the nuclear power plant, and the documentation of the Level 1 and 2 PSAs.

It is suggested that the regulatory authority agree with the utility on the format and content of the PSA documentation before the start of the Level 2 PSA and this needs to contain sufficient detail to allow the analysis to be traced. One suggestion for the way that the analysis could be documented is given in Table XXI of Ref. [6]. The first task of the review team would be to check that sufficient information has been provided to allow the review to proceed.

The analysis would normally relate to a frozen design as of an agreed date for a plant during the design stage or the actual design and operation for an existing plant. Where the PSA is being carried out as part of a periodic safety review, the regulatory authority might accept that the analysis could relate to the state of the plant after any proposed modifications have been completed.

Where the Level 2 PSA has been carried out as a follow-up to an existing Level 1 PSA, the documentation has to describe how the information in the Level 1 PSA, which is necessary to evaluate containment performance and the transport of radionuclides, has been transferred to the Level 2 PSA.

The analysis and documentation need to be in a form that can be easily amended to take account of new research and improved models as they become available.

2.2.4. Setting up the review team

The review team set up will have to be experienced in the techniques for carrying out best practice Level 2 PSAs. Although the basic framework and methods of the Level 2 PSA have been established, the review requires high levels of expertise and technical resources, which are different from those, required for the Level 1 PSA. The range of expertise needs to be sufficient to address all the issues that are likely to arise during the review of a Level 2 PSA and has to include the following:

- **systems analysts** who are familiar with the Level 1 PSA and the design of safety systems to address the Level 1/Level 2 PSA interface and the containment systems,
- **staff with an operating background** who are familiar with the Emergency Operating Procedures including the accident management measures for severe accidents,
- **experts in the severe accident phenomena** that could occur during and following core melt. This would include the physical and chemical processes that govern accident progression and determine the loads on the containment, and the way that radioactive material is transported from the molten fuel to the environment. It would also include expertise in the computer codes which are used to model severe accidents,
- **structural specialists** to address the performance of the containment following the loadings imposed by the severe accident and the failure modes that could occur, and
- **PSA specialists** to address the probabilistic quantification of the event trees developed to model the severe accident sequences that could occur and the associated uncertainties.

The review team is constituted in such a way as to carry out the review to the extent intended by the regulatory authority as discussed above. This could involve the use of external consultants to provide particular expertise not available within the regulatory authority. Additional training should be provided where necessary.

After the review has been completed, it is suggested that the regulatory authority retain a sufficient level of expertise to be able to review any of the uses being made of the Level 2 PSA.

Good interfaces are established between the review team and the PSA team to allow the free exchange of documentation and easy discussions. However, care has to be taken to ensure that the independence of the regulatory authority is not compromised.

2.2.5. Agreement on methods and identification of important issues

The methods that are available for performing Level 2 PSAs are discussed in detail in Section 2.4.

The usual way to model the progression of a severe accident in the Level 2 PSA is to use some form of event tree analysis — referred to as Containment Event Trees (CETs) or Accident Progression Event Trees (APETs). These vary significantly in terms of the number of nodes included in the model. These typically range from small event trees that have branch points representing different time regimes and some intermediate events, to complicated event trees that represent different time regimes, all major phenomena, system events and operator actions. Experience in carrying out Level 2 PSAs shows that both these approaches can be used to model the accident progression adequately — see Ref. [1].

The reviewer has to check that, in principle, the set of nodes chosen for the analysis is sufficient to model all the significant phenomena that could occur during the severe accident and provide the insights required by the aims and objectives that have been agreed for the analysis.

Where several methodologies are available to perform any portion of the analysis, it is important that the regulatory authority clearly point out to the PSA team which of these methodologies it would consider to be unacceptable. This will allow to avoid resources being used carrying out work that would later be considered inadequate.

The reviewer focuses on the areas of the Level 2 PSA that have the most significant impact on the results of the PSA. These are identified in Section 3.

2.2.6. Comparison with other PSAs

The review of the Level 2 PSA includes a comparison of the methods used and the results with PSAs which have been carried out for similar plants or plants with similar containment systems, where possible. It is the practice in many countries to use a previous, state of the art, PSA as a reference for the review of a new PSA. However, when doing this, differences between the plants need to be recognized very clearly.

2.2.7. Reworking of the analysis by the regulatory authority

The reviewers should consider whether there is a need to carry out any independent calculations or reworking of particular parts of the PSA to aid in the understanding of the PSA, and its sensitivities and uncertainties. The practice varies between the regulatory authorities in different countries. However, due to the complexity of the Level 2 PSA, this is not advised unless the regulatory body (or its consultants) has a sufficient level of expertise and resources.

2.2.8. Documentation of the review findings

The requirements for the documentation of the Level 2 PSA and the way to deal with recommendations arising out of the review are the same as those for the Level 1 PSA — see Ref. [5].

2.2.9. Interactions with the utility

The way that the interactions with the utility are conducted during the review of the Level 2 PSA are the same as those for the Level 1 PSA — see Ref. [5].

2.2.10. Research

The reviewers need to be aware of the extensive body of research which has been carried out in recent years and has provided a better understanding of the various phenomena that would occur during a severe accident. These have yielded experimental data and permitted computer code simulations of severe accident sequences and radiological releases and transportation.

In the course of the regulatory review of the Level 2 PSA, the reviewers may identify areas where further research would be worthwhile to provide a better understanding of the development of severe accidents, increase confidence in the analyses and reduce uncertainties. These areas should be put forward as topics suitable for national and international research programmes.

2.3. REVIEW OF THE AIMS, OBJECTIVES AND SCOPE OF THE LEVEL 2 PSA

It is suggested that the regulatory body agree on the aims, objectives and scope of the Level 2 PSA before the study is started. These aspects are discussed below.

2.3.1. Development of regulatory principles for the review of the PSA

As for the Level 1 PSA (see Ref. [5]), it is suggested that the regulatory authority set down the acceptance requirements which will be used to assess the acceptability of the Level 2 PSA and make these clear to the utility. This would normally require that the Level 2 PSA carried out is consistent with national guidance and methods as set out in Refs [1, 2, 7,] etc.

2.3.2. Aims and objectives of the PSA

As for Level 1 PSA (see Ref. [5]), it is suggested that the aims and objectives of the Level 2 PSA be agreed between the regulatory authority and the utility. It is important to understand what they are since, as pointed out in Ref. [4], “the review of a Level 2 PSA that is intended only to show that a nuclear power plant fulfils quantitative safety goals will be different than the review of a Level 2 PSA in which the objective is to produce information about the relative importance of systems and phenomena for accident management decisions or other purposes”.

For a typical Level 2 PSA, the overall aims would be to demonstrate that the plant has some inherent ability to withstand severe accidents, and to allow weaknesses in the level of protection to be identified. This may be used to support decision making, for example on the development of accident management measures, such as:

- depressurization of the primary circuit to prevent high pressure melt ejection,
- adding water to the containment to enhance heat removal from molten fuel using available systems such as the fire spray system, and
- use of portable equipment such as pumps or electrical generators to carry out safety functions.

or, the addition of further safety systems such as:

- hydrogen recombiners with sufficient capacity to deal with the rate of hydrogen generation following a severe accident,
- a filtered containment venting system which could be operated in the longer term to prevent containment failure due to overpressurization, or
- a core catcher or a core spreading area underneath the reactor.

In countries where risk targets or other criteria have been specified which relate to releases of radioactivity from the plant (either formal or informal), one of the aims of the PSA should be to provide the information that allows a comparison to be made with these risk criteria.

In setting the aims and objectives of the analysis, specific consideration needs to be given to the uncertainties in modelling the phenomena that would occur during a severe accident. Overall, the aim is to produce a *best estimate* model of the behaviour of the plant, which is not unduly distorted by introducing conservatism into the analysis (see Section 2.4.5).

2.3.3. Scope and applications of the PSA

As for the Level 1 PSA (see Ref. [5]), it is suggested that the regulatory authority and the utility agree on the scope and uses of the Level 2 PSA and ensure that this is sufficient to meet the aims and objectives of the analysis. If the scope of the PSA falls short of what would be expected, this should be brought to the attention of the utility so that the scope of the analysis can be changed at an early stage.

Agreement on the scope of the PSA is important since different end uses place different emphasis on the various parts of the analysis. For example, an analysis that was intended to look at hydrogen control or the ability of the containment to withstand the loadings that would arise during a severe accident might not need to address the transport of radioactive material within the containment.

The scope of the Level 2 PSA could range from a full scope analysis, which is part of a fully integrated Level 3 PSA, to a limited analysis. The latter could include an analysis which addresses the performance of the containment in severe accident situations but does not go on to determine the frequency and magnitude of the source terms that would arise from containment failure.

The best option is where the Level 2 PSA is part of a fully integrated PSA since the requirements for the Level 2 PSA will be recognized in carrying out the Level 1 PSA so that all plant related features that are important for the severe accident modelling will be fed into the Level 1 analysis. If this is not the case, the reviewers will need to pay special attention to the grouping/regrouping of the core melt sequences into plant damage states to ensure that the containment systems have been addressed correctly.

The agreement on the scope of the Level 2 PSA also needs to consider the following:

- the basic approach to the modelling of the progression of severe accidents (for example, using small or large event trees),
- how accident management measures and recovery actions are to be taken into account in the PSA,
- the range of sensitivity studies that need to be carried out (data, modelling assumptions),
- how the uncertainties in the severe accident modelling will be addressed and whether a full uncertainty analysis is required, and
- whether the analysis is to be extended to a Level 3 PSA.

It is suggested that the regulatory authority and the utility agree on the intended (and potential future) uses of the Level 2 PSA, and confirm that the proposed scope of the analysis is consistent with these uses. Again, if the intended uses do not meet the expectations of the regulatory authority, this needs to be brought to the attention of the utility at an early stage so that additional uses can be considered.

Some typical uses of the Level 2 PSA (taken from Ref. [6]) are as follows:

- to gain qualitative insights into the progression of severe accidents and containment performance,
- to identify plant specific vulnerabilities of the containment to severe accidents,
- to provide a basis for the resolution of specific regulatory concerns,
- to provide a basis for the demonstration of conformance with quantitative safety criteria,
- to identify major containment failure modes and to estimate the corresponding releases of radionuclides,
- to provide a basis for the evaluation of off-site emergency planning strategies,
- to evaluate the impacts of various uncertainties, including assumptions relating to phenomena, systems and modelling,
- to provide a basis for the development of plant specific accident management strategies,
- to provide a basis for plant specific backfit analysis and evaluation of risk reduction options,
- to provide a basis for the prioritization of research activities for minimization of risk significant uncertainties, and
- to provide a basis for a Level 3 PSA consistent with the PSA objectives.

In addition, the regulatory authority needs to consider what role the Level 2 PSA will play in the decision making process. If it is intended to use the insights gained from the Level 2 PSA as part of a risk informed approach, this should be taken into account in reaching the agreement about the uses of the PSA.

2.3.4. Sensitivity studies and uncertainty analysis

The review needs to verify that studies have been carried out to determine the extent to which results of the analysis are sensitive to:

- assumptions made in various portions of the analysis,
- analytical models selected (or the parameters that influence them) for severe accident phenomena, and
- data used in quantitative analysis.

In particular, the review verifies that the scope and level of detail of such studies are consistent with the objectives of the Level 2 PSA. For example, a structured sensitivity study addressing major assumptions, modelling parameters and data may be sufficient for studies in which the major aim of the study is to gain qualitative insights on plant response to severe accident conditions. A rigorous propagation of uncertainties may be necessary for studies in which the quantitative results are important (e.g. studies performed to demonstrate conformance to quantitative safety objectives).

In all cases, the reviewer has to verify that the sensitivity/uncertainty analyses address the topics in which there is significant uncertainty, and those that are dominant contributors to severe accident progression. Further guidance on this subject is given in Sections 3.3.5, 3.6.3, and 3.7.4.

2.4. REVIEW OF METHODS AND ASSUMPTIONS

2.4.1. State of the art

As for Level 1 PSA (see Ref. [5]), it is suggested that the regulatory authority and the utility agree on what the state of the art is for Level 2 PSA. In general terms, this would be expected to conform to the best modern practices as defined in Refs. [1], [2] and [5].

However, it is recognized that Level 2 PSA methods are developing and it is important that the methodology adopted reflects current trends where this has been shown to bring improvements over previous methods.

2.4.2. Level of detail

It is necessary for the reviewers to confirm that the level of detail proposed for the Level 2 PSA is appropriate for the applications, and sufficient to address significant interdependencies.

For example, plant specific calculations of severe accident behaviour are essential to an analysis performed for the purposes of measuring reductions in risk associated with proposed accident management measures; extensive use of 'reference plant' results is inappropriate such an application.

Interdependencies can arise in a number of ways, including the following:

- **support systems** such as electrical power systems and cooling water systems. Although they will have been included in the Level 1 PSA/Plant Damage States, their status is important in determining how the accident sequences progresses after core melt has occurred,
- **phenomena which are addressed in different time frames** in the model of how the accident progresses. For example, the likelihood of whether a hydrogen burn will occur in one time frame will depend on what safety systems have operated and whether a burn has occurred in an earlier time frame, and
- **human actions** which have been addressed in one time frame and may arise again in a later time frame.

2.4.3. Methods of analysis

It is necessary for the reviewers to determine whether the methods used for the analysis are adequate to meet the aims and objectives of the PSA. This would include:

- the codes used to model the progression of the severe accident,
- the framework (usually an event tree analysis) for the modelling of the severe accident sequences, and
- the probabilistic quantification of the event sequences.

More detailed guidance is given in Section 3 on the various aspects of the Level 2 PSA analyses.

The reviewers should identify what methods and tools are used (or proposed for use) for each of the Level 2 PSA tasks, and ensure that the ones to be used in the analysis are consistent with the state of the art. For uncertain sensitive areas of the analysis state of the art methods should be used. The review to check that the methods and tools have been correctly applied is described in Section 3.

2.4.4. Sources of data including expert judgement

The accident progression analysis carried out as part of the Level 2 PSA is usually done in an event tree framework referred to as Containment Event Trees (CETs) or Accident Progression Event Trees (APETs). These event trees delineate the various ways that an accident sequence can proceed after the onset of core damage.

Quantification of the event trees is accomplished by assigning conditional probabilities to each of the branches that emerge from event nodes in the trees.

Although the conditional probabilities for some of these branches can be quantified through the use of statistical data (e.g. those involving containment system operation or operator actions), many branches represent alternative outcomes of events that are governed by severe accident phenomena about which there is a significant degree of uncertainty. This uncertainty means that the physically correct outcome of the event is not known. Conditional probabilities are associated with such events to weight the outcomes according to the strength of the evidence suggesting one outcome versus another. These conditional probabilities are usually generated through a more or less formal expert judgement process.

The reviewers need to confirm that the framework for making these expert judgements is sound, is applied consistently throughout the analysis, and that the technical information used to make such judgements is stated and shown to be valid, as far as possible. This should take account of plant specific accident progression analysis that has been carried out, adaptation of analysis for other similar plants and applicable research data. Additional guidance on reviewing the assignment of branch point probabilities is provided in Section 3.6; further background on the use of expert judgement is given in Section 6.2 of Ref. [1].

2.4.5. Use of best estimate methods, assumptions and data

The PSA as a whole is based on best estimate methods, assumptions and data wherever possible. This is a particular requirement for the Level 2 PSA where conservative assumptions will lead to a model of the accident sequence progression which is not realistic, and hence may provide limited or misleading insights into where the weaknesses might be in the design and operation of the plant and which accident management measures would be useful in reducing the risk.

The reviewers should note that the effect of including conservatism in the Level 2 PSA may be significantly different from that in the Level 1 PSA. In the Level 1 PSA, the use of conservative safety systems success criteria, initiating event frequencies or component failure data would lead to an overestimate of the core damage frequency. However, in the Level 2 PSA, a conservative assumption in the modelling of one of the phenomena, which would occur during the severe accident, may not be conservative with respect to other phenomena.

Hence, it is important that the reviewers check that any conservatism included in the analysis would not lead to an unacceptable bias and distortion in the results of the PSA. This will be largely a matter of judgement on the part the reviewers.

Where an uncertainty analysis is performed, the values characterizing the input distributions (for example, median and error factors) always have to be realistically estimated.

2.4.6. Validation and verification of computer codes

The computer codes required for a Level 2 PSA include the codes which model the severe accident phenomenology, including the codes which model individual phenomena as well as the integrated codes (see Section 3.3), and the probabilistic codes for quantifying the events trees used to model the progression of the severe accident — see Section 3.6.

As for the computer codes used in the Level 1 PSA, those used in the Level 2 PSA also need to be validated and verified. In this context, **validation** is defined as providing the theoretical examination to demonstrate that the calculational methods used in the computer code are fit for the intended purpose. This may also involve comparison with experimental evidence. **Verification** is defined as ensuring that the controlling physical and logical equations have been correctly translated into computer code.

The reviewers need to check that the analysts have used the codes within their limits of applicability. In addition, they need to confirm that the predictions of the codes are consistent with the analysis carried out for similar plants and experimental information. Where integrated codes are used, their predictions are compared with those obtained using separate effects codes.

It is necessary for the reviewers to determine whether the codes which have been selected by the PSA team are fit for the intended purpose and that the users of the codes are experienced in their use and fully understand their limitations. It is suggested that the regulatory authority and the utility reach an agreement on the set of codes to be used.

2.5. REVIEW/AUDIT OF THE UTILITY'S PSA PRODUCTION PROCESS

2.5.1. Scope of the review/audit

As for the Level 1 PSA (see Ref. [5]), it is recommended that the regulatory authority should perform a review/audit of the process and the procedures being used by the utility to carry out the Level 2 PSA to give confidence that those parts of the PSA which have not been reviewed in detail have been performed satisfactorily.

For the Level 2 PSA, the aim would be to verify that the procedures used for each of the main PSA tasks addressed in Section 3 set out the basic principles and methodologies to be followed and are adequate to produce a PSA that fully meets its purpose.

The reviewers should check that the procedures are detailed enough to avoid misinterpretations by different members of the PSA team, so that they will be applied in a uniform and appropriate way throughout the PSA performance process, and will avoid the performance of tasks in a way that would not be acceptable.

In particular, regarding the users of the codes, the audit has to confirm that:

- the users are experienced in the use of the codes and understand the limitations,
- adequate guidance and training has been provided in the use of the codes, and
- the codes have been used to evaluate standard problems to gain experience.

2.5.2. Quality assurance

As for the Level 1 PSA, it is necessary that the reviewers of the Level 2 PSA determine whether the utility has QA programme in place for the performance of the PSA. This has to include arrangements for the conduct of an independent peer review.

2.5.3. Organization of the PSA production team

The review of the organization of the PSA production team for the Level 2 PSA addresses the same issues as for the Level 1 PSA (see Ref. [5]).

This review determines whether the team carrying out the Level 2 PSA has sufficient depth and breadth of experience in the issues addressed in the Level 2 analysis to enable the successful performance of the PSA.

2.5.4. Future updating/development of the PSA

It is suggested that the reviewers check that the PSA is being produced and documented in a way that makes it easy to update and to extend its use to other applications. The PSA report should be a living document, which is modified to incorporate any changes that result from the regulatory review, changes to the design or operation of the plant and changes in modelling assumptions or data.

The reviewers may consider it necessary to check that the utility has taken steps to maintain control of all the documents and workbooks used in the performance of the PSA, according to applicable QA requirements, to allow for any later audit or review by the regulatory authority.

It is considered good practice for the utility to maintain at least an adequate number of specialists on PSA on its staff to ensure the maintenance of the basic PSA capabilities acquired in the process of performance of the PSA. This group is a key element in the potential application of the PSA after it is completed.

3. CONDUCTING THE REVIEW OF THE LEVEL 2 PSA

This section provides guidance on the technical issues that need to be addressed in carrying out the review of a Level 2 PSA. This covers the following tasks:

- familiarization with plant data and systems,
- review of the Level 1 – Level 2 interface,
- familiarization with accident progression models used in the PSA,
- review of accident progression models,
- review of the containment performance analysis,
- review of the probabilistic modelling framework,
- review of the event quantification,
- review of characterization of source terms,
- review of treatment of uncertainties and of the basis for quantification of uncertain issues,
- review of integrated risk results, and
- interpretation and use of the results of the Level 2 PSA analysis.

Many of the examples used in this section are based on experience with performing reviews for Level 2 PSA of NPPs with PWRs.

3.1. FAMILIARIZATION WITH PLANT DATA AND SYSTEMS

The first task of the review process is to familiarize the review team with the plant, and plant response to potential severe accidents and severe accident phenomenology. This process is essential for the review team to identify characteristics that may influence the Level 2 PSA results, and may identify potential vulnerabilities in the containment design, prior to actually reviewing the analysis. This task consists of two sub-tasks: (a) familiarization with the design and operation of systems which may be initiated during a severe accident to mitigate its consequences and (b) collection/review of important plant and containment characteristics which may provide insights on accident progression and potential vulnerabilities.

3.1.1. Familiarization with systems which may be operated during a severe accident

The function and operation/actuation of plant systems need to be understood by the review team. Some of the relevant information may be readily available in the Level 2 PSA documents, but can be found in the Level 1 PSA documents. Typical information to be reviewed consists of:

- up to date systems P & IDs,
- system capacity, operating limits and actuation criteria, and
- ancillary information on the support systems which are needed by the primary or containment systems to operate.

The systems of relevance to potential mitigation/exacerbation of accident consequences are:

- all high and low pressure emergency core cooling system (ECCS),
- accumulators (for PWRs),
- reactor coolant system (RCS) depressurization systems,
- boration systems,

- long term heat removal systems (both for reactor and containment),
- containment isolation system,
- systems with a potential for containment bypass (interfaces between high and low pressure systems, letdown lines (PWRs),
- containment sprays,
- containment fan coolers,
- hydrogen control systems,
- containment venting systems,
- alternate RPV injection systems,
- moderator system (CANDU),
- alternate containment injection systems, and
- reactor building ventilation systems (BWRs).

A thorough review of the containment isolation system, and of other systems with a potential for containment bypass, is recommended.

If containment systems analysis is part of the Level 2 PSA, the procedures for the review are the same as described in Ref. [5]. Systems dependencies are of paramount importance. For instance, analysis of the containment isolation system is normally not part of a Level 1 study. This system is dependent on the availability of power sources (AC and/or DC), thus these dependencies must be clearly identified in the review. The same is true of the active hydrogen control systems, of the venting system, and of the containment cooling systems.

Level 2 PSAs may credit post-core damage operator interventions to mitigate the consequences of a severe accident — see Ref. [1]. In addition, systems may automatically initiate, if physical conditions change during the progression of an accident after core damage. For example, ECCS may actuate when available, if during a high-pressure transient some mechanism causes depressurization of the primary system. Therefore, the Emergency Operating Procedures (EOPs), must also be checked, to understand operator response in case of a severe accident, before and after core damage, and the potential for interventions using available systems after core damage. The degree to which these procedures are supported by training and exercises is important in assessing the probability that these procedures may be carried out successfully.

3.1.2. Plant and containment data

A useful way for the review team to develop a general understanding of plant characteristics is to compare key design and operating parameters for the plant being analysed with those of plants similar in design and configuration. This information can also suggest 'typical' severe accident vulnerabilities that have to be addressed in the Level 2 PSA (see Refs. [1], [6]). Collecting and evaluating data for key plant and containment design features is, therefore, a critical part of the review process. Example plant and containment features are listed in Table I; possible uses of this information are suggested in the right-hand column.

No amount of data or drawings can substitute for the visual images a reviewer obtains by actually seeing the systems being analysed. Consequently, a plant walk-down of the containment and key plant systems is strongly recommended for each member of the review team.

TABLE I. SAMPLE OF KEY PLANT/CONTAINMENT DESIGN CHARACTERISTICS

Key plant/containment design feature	Potential uses and other comments
Reactor type (BWR, PWR, other)	Identify other similar plants
Power level	Rated power establishes overall plant size. ATWS
Fuel, cladding type and mix	Oxide or mixed oxide fuels; zirconium or stainless steel cladding and other materials have their associated peak core temperature limits, melting characteristics, hydrogen generation rates, and concrete interaction behaviour
RCS coolant/moderator volume	Reactor vessel and coolant piping coolant inventories can be used to estimate maximum depths of water on the containment floor.
Number & coolant volume of accumulator(s)	Passive core/debris cooling capability
Mass of coolant available and maximum pressure for ECCS	Long term cooling capability, and extent of depressurization required for using low-pressure systems
Containment free volume	Distribution of volume [drywell vs. wetwell (BWR), above vs. below operating deck (PWR)] suggests potential for non-condensable gas build up and hydrogen concentration
Containment design pressure/temperature	Capacity to withstand quasi-static loads
Containment structure	Steel shell, concrete, etc. suggests appropriate failure modes
Mass of fuel	Total energy content of core
Mass of cladding material	Indicator of maximum hydrogen production
Control rod materials and mass	Low-temperature melting material
Key plant/containment design feature	Potential uses and other comments
Suppression pool volume (BWR)	Scaling parameter for loss of decay heat removal sequences
Suppression tower (WWER)	Scaling parameter for loss of decay heat removal sequences
Concrete composition	Non-condensable gas generation due to core-concrete interaction, after vessel failure

TABLE I. (cont.)

Cavity/pedestal design	Suggests potential for debris dispersal during high-pressure sequences, and ex-vessel debris/structure interactions during low-pressure sequences
Sump(s), volume and location (PWR)	Possibility of degraded recirculation cooling due to clogging with debris
Containment geometry	Extent of compartmentalization suggests potential for local combustible gas accumulation
Reactor power/RCS volume ratio	Estimate accident progression times, recovery opportunities.

3.2. LEVEL 1–LEVEL 2 INTERFACE

3.2.1. Plant damage states

The manner in which Level 1 PSA results are carried forward to the Level 2 analysis should be reviewed. The interface between the two studies is most often accomplished through the definition of 'Plant Damage States' which define the initial and boundary conditions necessary for conducting severe accident progression analysis.

The Level 1–2 interface, or Plant Damage State (PDS) analysis, may be performed at the conclusion of the Level 1 PSA, and can be reviewed as a product of the study, as described in Ref. [5]. Often, however, the interface is developed at a later time, as an initial step to a Level 2 PSA. Therefore, the discussion given in Ref. [5] is partially repeated here and extended. Extension of the discussion is necessary because many containment systems are usually beyond the scope of Level 1 studies, thus the status of these systems may not be identifiable from the Level 1 PSA models. In this case, the availability of containment systems during various core damage sequences must be addressed by means of an extension to the Level 1 system models. In some Level 2 PSAs, post core damage operator interventions are also identified in the definition of the plant damage states, as explained below.

3.2.2. Definition of plant damage states characteristics

The objective of the PDS analysis is to combine event sequences from the Level 1 accident analysis that result in similar severe accident progression, containment response, and fission product release to the environment. By doing so, the number of unique accident conditions that must be addressed in the Level 2 PSA is greatly reduced. For example, a Level 1 PSA would typically use different event trees to model core damage following a spurious reactor trip versus a loss of feedwater. However, from the point of view of containment response sequences from these two event trees may be similar and might be combined for the Level 2 PSA. It should also be noted that in some cases Level 1 sequences may be split between different PDS (rather than combined) since information such as containment system operation may not have been important for a Level 1 point of view and thus was not included in the Level 1 event trees. This aspect is referred to in more detail in a subsequent paragraph.

To accomplish the PDS grouping, the Level 1 results are sorted according to the physical state of plant systems that were demanded prior to the onset of core damage, and the availability of systems that could be actuated subsequent to core damage, thereby terminating the accident, or mitigating its consequences. It is necessary that the criteria used to combine similar core damage sequences be carefully reviewed to ensure that plant characteristics governing severe accident progression, containment response and fission product release to the environment are properly accounted for.

Typical grouping criteria, used for example for LWRs, include:

- the type of initiating event that has occurred (intact primary circuit or LOCA),
- the status of safety systems, such reactor protection system, residual heat removal system, and emergency core cooling (injection and/or recirculation),
- the availability of AC and DC power,
- the primary circuit pressure (high or low) at the time of core damage,
- the status of pressure reduction systems (e.g. ADS for BWRs, PORV position for PWRs),
- the time at which core damage occurs (early or late relative to the time of reactor scram),
- the integrity of the containment (intact, failed, isolation failure, bypassed due to a SGTR or an interfacing systems LOCA),
- suppression systems status when core damage occurs, and
- the availability of the containment protection systems (containment sprays, heat removal systems, hydrogen mixing/recombiners/ignitors).

The reviewer should be cognizant of the fact that for many accident sequences, the status of particular systems may not be known directly from Level 1 system models. For example, large break LOCA success criteria may require at least one of the (PWR) accumulators to function to prevent core damage. For event sequences involving failure of all accumulators, the Level 1 accident sequence event trees would not need to address the operation of other ECCS systems, and the sequence would proceed directly to core damage. However, the Level 2 analysis would need to know whether high- and/or low-pressure coolant injection systems were available during the sequence. As described in Section 3.2.3, determining the status of such systems, and other systems not addressed in the normal Level 1 event sequence models, requires an extension of the typical Level 1 PSA models. A review of the Level 1–2 interface, therefore, must determine how the Level 1 models were modified to capture such information in the PDS definitions.

Currently, EOPs for many operating plants include operator interventions when it is expected that accident conditions are irreversible, and core damage will very likely occur within a short period of time. These are not normally included in the Level 1 study.

For example, in the EOPs for some PWRs, the time for irreversible accident conditions is identified on the basis of elevated core exit temperatures, which is an indication of permanent inadequate core cooling. In the case of some BWRs, operators are instructed to initiate some procedures, which may not be normally considered in the Level 1 PSA, on the basis of low level in the reactor, or because of presence of hydrogen in containment, or because of high temperature in the pressure suppression pool. These may include actions such as emergency depressurization, feed and bleed, etc. All these actions can be considered part of accident management. However, the review team needs to check whether and to what extent these actions and the use of the pertinent systems have been considered already in the Level 1 PSA model.

Interventions which may be prescribed after core damage include:

- primary system depressurization after core damage,
- initiation of alternate core injection systems,
- flooding the containment,
- flooding the reactor cavity/pedestal,
- venting the containment,
- venting the reactor pressure vessel (BWRs),
- refilling the steam generators (PWRs),
- actuation of the hydrogen control systems,
- actuation of containment sprays from alternate injection systems.

Therefore, the Level 1–2 interface may include more information than shown in the list of grouping criteria and as recommended in Ref. [6]. For instance, definition of primary system pressure *before vessel breach*, rather than pressure *at the onset of core damage* might be more appropriate when post-core-damage operator actions have been incorporated in the analysis.

3.2.3. Plant damage states analysis and quantification

The systems availability aspect of the PDS definitions can be addressed in several ways. One is to extend the Level 1 event trees to include top events addressing the availability of the containment systems, so that their system fault trees can be linked and dependencies accounted for in the evaluation. Another way is to model all the systems (containment and other mitigative systems) in the Containment Event Trees, although care is then needed to ensure that correlations with the Level 1 sequences, such as dependencies on common support systems, is maintained. Yet another way is to use a separate computer programme which takes the cut set equation information from the Level 1 event trees, links in the fault trees for the containment systems and, if appropriate, for the accident management systems, and acts essentially as an extension to the Level 1 trees (bridge trees). Such a programme can also be written to group the sequences according to all of the characteristics in the definitions of the PDSs, with input of the appropriate information on timing, pressure etc., giving the frequency of each PDS as output, ready for the Level 2 analysis. Where this approach is taken, the reviewers are recommended to check that the assumptions, simplifications and dependencies have been clearly described (see Section 3.1).

For bridge trees that include fault trees of systems not included in Level 1, it is necessary that the system reliability models be reviewed as described in Ref. [5].

3.2.4. Human reliability analysis (HRA) related to plant damage states

Analyses are performed to quantify the PDSs address operator actions after the onset of core damage, the manner in which human errors associated with these actions are addressed need to be reviewed. At a minimum, the evaluation of post-core-damage human error rates considers prior operator performance and dependencies. The HRA also appreciates the levels of stress for personnel and uncertainties in the availability of reliable indications and signals in a severe accident environment.

It is suggested that the Level 2 HRA be reviewed according to the guidelines presented in the IAEA-TECDOC-1135 [5]. When using the guidelines, the features specific to Level 2 may be emphasized especially regarding:

- staffing,
- decision making, and
- severe plant conditions.

Staffing

The review may emphasize aspects related to the personnel involved in Level 2 actions. Usually, a crisis team (CT), separate from the control room shift, is responsible for decision making. An adequate HRA would therefore make concrete references to:

- the plant documents, such as the organizational handbook, where the role of the CT is described;
- the manner of notifying the CT together with supporting staff, such as fire brigade, including the expected arrival times and the related exercises.

The review may spot-check such information by referring to the organizational handbook or by contacting the responsible plant staff.

If the plant does not have a CT organization or if the HRA does not document the information outlined above, taking credit of Level 2 actions would become an issue deserving special attention. The human error probability (HEP) assessments of important actions is then reviewed with special care.

If the CT is responsible for decision making, the review may quickly spot-check the scheduled CT arrival time against the time windows of the credited Level 2 actions.

Decision making

Uncertainties increase when shifting from Level 1 to Level 2 scenarios. Regarding HRA, these uncertainties may concern the process of decision making. Less explicit decision rule, such as *if* <pattern of indications> *then* <action> required, are available [8]. However, the HRA needs to refer to procedural rules *or* to trained rules that are supposed to support the decisions of taking the credited actions.

In any case, the quantification of decision making in Level 2 scenarios points to limitations of current HEP assessment techniques. It is expected therefore that the HRA carries out the quantification with a reasonable amount of conservatism.

Severe plant conditions

Level 2 HEPs should account for severe plant conditions. For a selected set of important actions, the review may spot-check the following HEP impacts[1]:

- dependency from preceding (Level 1) action failures,

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1 For selected issues or subjects, the reviewer may refer to the verification procedures presented in Ref. [8].

- dependency from preceding (Level 1) action failures,
- preceding equipment failures that may disable a Level 2 action (for the review it would be helpful to have a list of the equipment (including instrumentation) needed per action),
- inaccessibility of performance locations (for the review it would be helpful to know where the credited actions are supposed to take place),
- increased stress/workload (it is expected that stress in Level 2 scenarios is higher than in Level 1 scenarios).

3.2.5. Results of the PDS analysis

All core damage event sequences are be assigned to a PDS, and the sum of the PDS frequencies should be approximately equal to the total core damage frequency (e.g. cut-off criteria should be sufficiently low).

In some PSAs, event sequences (or minimal cut sets) with very low frequency are ignored in the PDS grouping process. If a cut-off frequency is applied in a PSA, the reviewer needs to check:

(a) that the total frequency of event sequences below the cut off value is a small fraction of the total core damage frequency (e.g. less than 1 %); and
(b) accident sequences that could potentially lead to large consequences (i.e. containment bypass sequences, steam generator tube rupture accidents, sequences with containment isolation failure) are not systematically removed from the PDS process.

3.3. ACCIDENT PROGRESSION MODELS

3.3.1. Accident progression models

Deterministic analysis of reactor and containment behaviour during postulated accident sequences represent the principal basis for phenomenological event quantification in a Level 2 PSA. Such analyses provide a plant specific technical basis for distinguishing the phenomenological event branch probabilities. The probabilistic framework of a Level 2 PSA (discussed in Section 3.5) is the mechanism for delineating and quantifying uncertainties in deterministic severe accident analyses. This section outlines various features of deterministic accident progression models that are examined in the course of a Level 2 PSA review.

3.3.2. Identify computer codes used to perform accident progression analysis

The reviewer identifies the computational tools used to perform accident progression calculations. In some studies, a single integrated severe accident analysis computer code is used to model all aspects of the severe accident progression, including:

- reactor coolant system thermal-hydraulic response (prior to the onset of core damage),
- core heat up, fuel degradation and material relocation within the reactor vessel,
- possible failure of the reactor vessel pressure boundary, and subsequent release of molten fuel and core debris to the containment,
- thermal and chemical interactions between core debris and containment structures, such as concrete (or steel) floors and walls, pools of water and the containment atmosphere, and
- containment behaviour (including its pressure/temperature history, hydrogen mixing and combustion, and the effect of the operation of containment safeguard systems).

Computer codes that address the entire spectrum of processes include MAAP, MELCOR, ESCADRE and THALES-2. Consequently, these codes provide an integrated framework for evaluating the timing of key accident events, thermodynamic histories of the reactor coolant system, core and containment, and corresponding estimates of fission product release and transport. However, the broad scope of these codes (and the requirement that they complete calculations in a reasonably short time), demands simplifications in many aspects of accident progression models. Examples of these simplifications include: lumped parameter approximations to material transport and thermodynamic conservation equations, and the use of empirical correlations for complex physical processes. The reviewer needs to be aware of the areas in which these simplifications are applied, and determine whether their effects are taken into account in the Level 2 PSA. The manner in which these effects (and other modelling uncertainties) are considered is addressed in more detail in Section 3.6.

In some studies, calculations with the integrated computer codes described above are replaced by, or supplemented with, calculations performed with other computer codes that address specific aspects of severe accident progression. Examples of such computer codes are listed in Table I. In general, the narrower scope of these codes allows them to address important accident phenomena in a greater level of detail than is afforded by the integrated computer codes. The reviewer should take note of the specific areas in which these codes are used, and determine whether results obtained with them are used in conjunction with, or in place of, those obtained from integral code calculations. A list of specific accident phenomena to consider in this exercise is given in the next Section.

TABLE II. SEVERE ACCIDENT COMPUTER CODES

		In-Vessel Phenomena				
Computer Code	Ref.	Thermal-hydraulics	Core degrad-ation	Fission product release from fuel	Fission product transport in RCS	Reactor vessel failure
ART	9			X	X	
ATHLET-CD	10	X	X	X	X	X
BWRSAR	11	X	X			X
CATHARE	12	X				
ESCADRE	13	X	X	X	X	X
ESTER	14	X	X	X	X	
ICARE	15	X	X	X		
IFCI	16					X (FCI)[a]
MAAP	17	X	X	X	X	X
MELCOR	18	X	X	X	X	X
PM-ALPHA/EPROSE	19					X (FCI)
SCDAP-RELAP5	20	X	X	X	X	X
STCP	21	X	X	X	X	X
TEXAS	22					X (FCI)
THALES-2	23	X	X	X	X	X
VICTORIA	24			X	X	

[a] FCI: Fuel-coolant interactions (i.e. steam explosions).

TABLE II. (cont.)

		Ex-Vessel Phenomena				
Computer Code	Ref.	Core-concrete interaction	Fission product release from core debris	Fission product transport in containment	Hydrogen combust-ion	Cont-ainment response
CONTAIN	25	X	X	X	X	X
CORCON/ MOD3	26	X				
ESCADRE	13	X	X	X	X	X
FIPLOC	27			X		
HECTR	28				X	
HMS	29				X	
MAAP	17	X	X	X	X	X
MELCOR	18	X	X	X	X	X
RALOC	30				X	X
STCP	21	X	X	X	X	X
THALES-2	23	X	X	X	X	X
WECHSL	31	X	X			

3.3.3. Account for treatment of important accident phenomena

A thorough review of a Level 2 PSA includes a check to ensure important accident phenomena are addressed by plant specific analysis (e.g. included as an element of computer code calculations), or by application of information from other credible and relevant sources (e.g. experiments or published ‘reference’ plant analysis). Table III provides a suggested list of accident phenomena to include in this check. For each item in the list, the reviewer should be able to identify the model (e.g. computer code) or data source used to address it (the list may be different, according to reactor type; e.g. many of the items in Table III may not apply to RBMK reactors, while some may be missing).

If published data from experiments or reference plant analysis is used to evaluate certain phenomena, the relevance of that information to the plant being studied needs to be confirmed. If plant specific analysis is performed (using one of the computer code listed in Section 3.3.1), the data used to perform the calculations is checked as described in the next Section.

TABLE III. ACCIDENT PHENOMENA TO BE ADDRESSED WITH ACCIDENT PROGRESSION MODELS

Time domain	Phenomena
RCS thermal-hydraulic behaviour prior to core damage	Depletion of primary coolant inventory Temporal changes in core power Reduction in reactor vessel water level Thermodynamic effects of steam generator, relief valve, and coolant injection system operation (along with other systems represented in Plant Damage State definitions). Asymmetric RCS coolant flow and heat transfer associated with pipe breaks (LOCAs), pressurizer behaviour, or non-uniform steam generator operation
In-vessel core degradation	Fuel heat up, and heat transfer to neighbouring structures Metal-water reactions and accompanying hydrogen generation Eutectic material formation and associated changes in thermo-physical properties Control material melting and relocation Clad ballooning, failure, melting and relocation Dissolution of fuel, and relocation with molten metals Re-freezing of previously molten material on cooler surfaces Formation of local and/or core-wide blockages Accumulation of molten materials above large scale blockages Enhanced steam/hydrogen generation accompanying the introduction water to core debris (e.g. from mid-period accumulator operation) Structural collapse of fuel rods (formation of particulate) and other structures Relocation of molten material (via pour) and/or regional collapse of core into lower plenum of reactor vessel Quenching of core debris in the lower plenum and debris formation on lower head surface
RCS pressure boundary failure	Buoyancy-driven natural circulation flow within the reactor vessel Counter-current natural circulation flow within RCS piping and steam generators (PWRs) Heat transfer to the RCS pressure boundary including cumulative damage leading to creep-rupture (at locations such as hot legs nozzles, pressurizer surge line, and steam generator tubes)
Reactor vessel failure and debris relocation to containment	Energetic fuel-coolant interactions within reactor vessel lower head (alternative to quench), resulting in steam explosion Reheating of quenched core debris in lower head, and molten pool formation Cumulative thermal damage to reactor vessel lower head leading to creep rupture Local pour of molten material onto lower head surface leading to jet impingement, possible plugging and failure of lower head penetration Relocation of molten materials and particulate debris from lower head to containment floor In-vessel debris configuration and coolability

TABLE III. (cont.)

Time domain	Phenomena
Energetic phenomena accompanying vessel failure	High-pressure melt ejection, debris fragmentation and dispersal in the containment atmosphere Hydrogen generation, ignition and combustion Direct containment heating Energetic fuel-coolant interaction on containment floor and ex-vessel steam explosion Direct impingement of ejected core debris on thin (steel) containment boundary structures Reactor pressure vessel reaction forces and movement accompanying vessel failure
Ex-vessel behaviour of core debris (long term)	Corium-concrete interactions (non-condensable gas and steam generation, concrete ablation and accompanying changes to corium properties) Heat transfer and damage to containment pressure boundary due to direct debris contact Basemat penetration Ex-vessel debris configuration and coolability
Containment response	Steam and non-condensable gas accumulation and resulting changes in containment pressure Hydrogen stratification or mixing, as appropriate Thermodynamic effects of containment sprays, coolers, and pressure suppression system operation (along with other systems represented in Plant Damage State definitions) Ignition and burning of combustible gases (including diffusion flames, deflagrations and detonations, as appropriate) Containment failure due to over-pressure or over-temperature conditions

3.3.4. Review model input data

A large amount of input data from different sources needs to be reviewed when performing an evaluation of deterministic Level 2 PSA calculations. The input can be grouped as follows:

(a) ***Plant specific data used to represent the plant***. Basic data used to define the configuration, geometry and material composition of the plant has to be checked to ensure that these data are used appropriately in the models representing the plant. For example, the total volume of water (or other coolant) in the RCS and the secondary side of steam generators, the volumes of various compartments in containment and the means by which they communicate with each other, and type of concrete used to construct the containment needs to be verified by comparison to plant design documents. As described in Section 4, thorough documentation and independent verification of this type of information is found in quality assurance documents associated with the Level 2 PSA. If such documentation is not available, the reviewer will have to spot check values of key parameters covering various portions of the plant model by comparing them to plant design documents.

(b) ***Plant modelling structure (spatial nodalization schemes)***. The level of detail used to develop a nodal thermodynamic model (i.e. lumped parameter control volumes) has to be examined. This review includes RCS and containment nodalization schemes as well as the core nodalization structure. Ideally, model optimization studies would have been performed which indicate the sensitivity (if any) to alternative schemes for such models[2]. In the absence of such information, the reviewer should confirm that the spatial nodalization schemes used by the analysts are consistent with contemporary approaches used for other, similar plants.

Areas in which the plant model is asserted to be 'conservative' with respect to some process call for particular attention. For example, a model that neglects the heat capacity associated with boundary structures might be claimed to be conservative with respect to the calculation of peak internal atmosphere temperatures. However, such simplifications might adversely (i.e. non-conservatively) affect other coupled phenomena, such as steam condensation on walls, and lead to hydrogen stratification.

(c) ***Accident scenario input***. Input data used to define the characteristics of a specific accident sequence has to be verified. The specific relationship between a computer code calculation and the accident sequences (or plant damage states) it is supposed to represent needs to be noted and checked against the source(s) of data used for eventual quantification of events in the CET (see Section 3.6). Example parameters to be examined include:
- leak areas and their location,
- performance specifications for operating equipment and systems (e.g. actuation/termination criteria, number of operating trains, flow or energy exchange rates, etc.), and
- timing of assumed (successful) operator actions.

(d) ***Input for models of accident phenomena***. Unless otherwise required for reasons delineated in the accident analysis documentation, model input that controls how severe accident phenomena are treated is consistent from one calculation to another. Exceptions are sensitivity calculations performed with the explicit purpose of characterizing the effect(s) of alternative credible models for uncertain phenomena. The reviewer needs to verify that a *self-consistent* set of phenomenological modelling assumptions (i.e. code input) is used to generate the entire set of calculations for representing baseline accident behaviour. Calculations that are performed with modelling assumptions that differ from the baseline values have to be noted. The manner in which they are used in the Level 2 PSA will then be checked as described in Section 3.6.

As above, areas in which selected *modelling options* are asserted to be 'conservative' call for particular attention. For example, modelling choices that inhibit debris fragmentation and cooling in-vessel (which might be viewed as conservative from the point of view of thermal challenges to reactor vessel lower head integrity), also reduce steam production rates,

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[2] For example, sensitivity studies might have been performed in which the calculated core thermal response to a typical accident sequence was calculated using alternative axial and radial nodalization schemes; similarly, the effects of thermal-hydraulic modelling simplifications (such as the number of interconnected control volumes used to represent multiple, small compartments in the containment) may have been examined in sensitivity studies.

thereby decreasing in-vessel hydrogen generation. Such assertions are to be noted by the reviewer. The extent to which such views are carried forward into the probabilistic models (reviewed in Sections 3.5 and 3.6) need to be examined.

3.3.5. Review calculated results

It is usually impractical for a reviewer to examine the details of each and every calculation performed in support of a Level 2 PSA. However, it is necessary to ensure that results are (in general terms) consistent with contemporary analyses for other similar plants. For example, the open literature contains numerous reports of detailed severe accident calculations, performed with various computer codes, for accident sequences commonly found in Level 2 PSAs (e.g. station blackout, small break LOCAs, loss of decay heat removal). Comparisons of calculated results to such reference analyses provides a useful basis for gauging the extent to which unique plant design or operating characteristics influence severe accident progression. In the absence of such information (e.g. for unique plant designs), the reviewer may check global results by means of simple hand calculations; e.g. mass/energy balances to estimate the timing of key events.

3.3.6. Assessment of major uncertainties treatment

All engineering calculations are subject to some form of uncertainty. Although most Level 2 PSAs do not treat uncertainties in a rigorous manner, they should nevertheless be accounted for via structured sensitivity studies, or some other means. A well structured sensitivity analysis can identify which events and phenomena have the greatest impact on the calculated probability of containment failure, or the magnitude of fission product source terms, without estimating their uncertainties quantitatively (i.e. development of uncertainty distributions for all important output parameters). Without such sensitivity analyses, the Level 2 PSA may be considered as incomplete.

Typical issues that are examined as part of a structured sensitivity analysis are listed in Table IV. More extensive descriptions of major severe accident uncertainties can be found in Refs. [32, 33]. The specific parameters that can be varied to study the sensitivity of plant response to these issues depend strongly on the model (i.e. computer code) used for the analysis. However, most codes provide some flexibility to the analyst for performing meaningful sensitivity calculations.

3.4. CONTAINMENT PERFORMANCE ANALYSIS

Calculations of severe accident progression (discussed in Section 3.3) generate pressure and temperature histories within containment during various accident sequences. To determine whether the containment pressure boundary will be able to withstand these (and other) loads, quantitative estimates of its structural performance limits must be generated. Because challenges to containment integrity can take many forms, the analysis of containment performance limits must address several topics. Typically, the following containment challenges are considered in establishing containment performance limits:

- internal, slow quasi-static and rapid pressurization transients greater than nominal design conditions,
- high temperatures,

TABLE IV. TYPICAL ISSUES EXAMINED IN SENSITIVITY STUDIES

In-vessel accident phenomena

- Core debris relocation, fragmentation and coolability
- Steam availability and associated hydrogen generation
- Natural circulation (above the core) and induced RCS pressure boundary failures
- Debris coolability and configuration in the reactor vessel lower head
- Mode of reactor vessel failure
- Hydrogen generation

Ex-vessel core/debris phenomena

- Debris fragmentation and dispersal following vessel breach at high pressure (direct containment heating issues)
- Fuel/coolant interactions on the containment floor
- Debris coolability during corium-concrete interactions
- Non condensable gas generation

Containment performance

- Containment failure pressure, (particularly for concrete structures)
- Thermal degradation of containment penetration seals
- Leakage area associated with containment failure

Containment phenomena

- Heat loss to the environment for a steel-shell containment
- Natural circulation (buoyancy-driven) flows
- Hydrogen distribution (mixing/stratification)
- Hydrogen combustion (initiation/concentration threshold, burn completeness, flame propagation, speed)
- Effectiveness of engineered safety systems

Other

- Effect of operator actions

- thermo-mechanical erosion of concrete and steel structures (if contact with ejected core debris is possible),
- impact from internally-generated missiles, and
- localized dynamic loads, such as shock waves.

In some instances, these challenges may exist simultaneously. For example, high temperatures often accompany high pressures.

Engineering calculations of structural response to these types of challenges are performed as part of a complete Level 2 PSA. Quantitative failure criteria are developed as the primary reference for estimating the likelihood of containment failure for a wide spectrum of accident sequences. These criteria are based on plant specific design and construction data and represent realistic material response properties.

The reviewer checks that the following features of the containment pressure boundary are included in the analysis:

- containment configuration, construction materials and reinforcement (e.g. free-standing steel shell; concrete-backed steel shell; pre-stressed, post-tensioned or reinforced concrete),
- design of containment liner with regard to containment penetrations,
- penetrations of all sizes, their location in the containment structure and local reinforcement (e.g. equipment and personnel hatches, piping penetrations, electrical penetration assemblies, ventilation system penetrations),
- penetration seal configuration and materials, and
- local discontinuities in the containment structure (e.g. shape transitions, wall anchorage to floors, changes in steel shell or concrete reinforcement).

3.4.1. Structural response analysis

An analysis of containment structural response to imposed loads has to consider interactions between the containment structure and neighbouring structures, internal and external (e.g. reactor vessel and pedestal, auxiliary buildings, piping that penetrates the containment boundary).

It is recommended that, the analytical tools used to develop containment failure criteria be accepted industry standards (e.g. rigorous, finite element computer codes), or a method supported by experimental validation. Alternatively, experimental results can be used directly. For example, direct experimental data are available in the open literature regarding criteria for reactor containment penetration seal performance under conditions of high temperatures and pressure (see Ref. [34]).

A review of containment performance analysis also needs to examine the terms in which containment failure criteria are stated. A complete structural performance assessment distinguishes conditions that would result in catastrophic failure of the pressure boundary from those that result in more limited leakage, and identify the anticipated location of failure. For example, finite element analysis may suggest that quasi-static pressure increases at relatively low temperatures may lead to tearing of a cylindrical (PWR) containment wall where it joins the flat basemat floor. Under these conditions, and at this location, the anticipated size of resulting opening in the containment wall is expected to be large. At the same pressure, but significantly higher temperatures, finite element analysis may suggest a different failure mechanism, location, and as a result a different size.

If external events are considered in the PSA, containment structural response to postulated seismic events need to be reviewed. As with other mechanisms for containment failure, the relationship between seismic intensity (e.g. ground acceleration) and the location and size of containment failure are identified in the study. Analysis of structural response to dynamic loads (i.e. impulsive loads) is considerably more difficult than traditional static structural response analysis. Quantitative, plant specific failure criteria may not be practical to develop. Rather, information presented in the open literature is commonly used to treat the possibility of containment failure due to in-vessel steam explosions (see Ref. [35]), and catastrophic structural failure is often the assumed consequence of hydrogen detonations.

3.4.2. Containment bypass

In addition to structural failure of the containment pressure boundary, a thorough characterization of containment performance needs to examine mechanisms and pathways by which fission product released from the RCS may bypass the containment and be released directly to the environment. The reviewer has to examine analyses performed to identify locations, pathways and associated sizes of bypass mechanisms. Typical bypass mechanisms include:

- interfacing system LOCAs, and
- steam generator tube rupture (SGTR).

With regard to SGTR, the reviewer also needs to check that such events are not only treated as initiating events (carried forward from the Level 1 analysis), but are also considered as an event that may occur during in-vessel core degradation (see phenomena listed in Table III).

3.4.3. Failure of containment isolation

Two types of containment isolation failures are normally analysed and should be included in the Level 2 PSA. These are pre-existing leaks (i.e. undetected penetration seals failure, or isolation valves which are failed open), and consequential isolation failure paths (i.e. occurring after the initiating event).

Only leak paths that lead to leakage rates substantially higher than nominal, or the design basis rate need be considered.

3.5. PROBABILISTIC MODELLING FRAMEWORK

The primary function of a probabilistic model for evaluating containment performance, is to provide a structured framework for organising and displaying the alternative accident progressions that may evolve from a given core damage sequence, or a plant damage state. This framework generally takes the form of containment event trees or accident progression event trees. These logic structures are the backbone of the Level 2 PSA model, and have to be reviewed thoroughly.

3.5.1. Content and format of the Level 2 model

In reviewing the Level 2 probabilistic model, the following features need to be examined to ensure a comprehensive and scrutable assessment of containment performance:

Explicit recognition of the important time phases of severe accident progression. Different phenomena may control the nature and intensity of challenges to containment integrity and the release and transport of radionuclides as an accident proceeds in time. The following time frames should be identified in a Level 2 analysis:

- ***After the initiating event, but before the onset of core damage***. This time period establishes important initial conditions for containment response after core damage begins.
- ***After the core damage begins, but prior to failure of the reactor vessel lower head***. This period is characterized by core damage and radionuclide release from fuel while core material is confined within the reactor vessel.
- ***Immediately following reactor vessel failure***. Prior analysis of containment performance suggests that many of the important challenges to containment integrity occur just prior to or following reactor vessel failure. These challenges may be short lived, but often occur only as a direct consequence of the release of molten core materials from the reactor vessel immediately following lower head failure.
- ***Long term accident behaviour***. Some accident sequences evolve rather slowly and generate relatively benign loads to containment structures early in the accident progression. However, in the absence of some mechanism by which energy generated within the containment can be safely rejected to the environment, these loads may steadily increase to the point of failure in the long term.

When linked end to end, these time frames should provide a clear and chronological description of the alternative accident progressions represented in the PSA. The reviewer should be able to 'trace' individual accident sequences from the Level 1 PSA (perhaps via a plant damage state) through the alternative progressions of post-core damage accident behaviour.

Distinction of discrete system events from phenomena. Probabilities associated with 'events' in a containment event tree (or other type of logic model) are of at least two different types. One represents the conditional probability that an engineered system will operate or fail to operate upon demand or that a human will perform, or fail to perform a specific activity. The probabilities of such events directly parallel those represented in Level 1 PSA accident sequence event trees, and are developed in a similar manner. The other type represents uncertainty in the occurrence or effects of severe accident phenomena. For example, an 'event' may be included in a Level 2 PSA logic structure that depicts the divergence in plant behaviour that occurs when a hydrogen burn occurs, or does not occur at some point in time. In this case, the split fraction associated with this event is not based on reliability data. Rather, it is a reflection of the uncertainties in the engineering analyses required to characterize hydrogen generation, release, distribution and combustion. The reviewer checks to see that these distinct types of events are identified and treated appropriately in the logic.

Consistency in the treatment of severe accident events from one time frame to another. Many events or phenomena may occur during several different time frames of a severe accident. However, certain limitations apply to the composite (integral) contribution of some phenomena over the entire accident sequence and these are represented in the formulation of a probabilistic model.

A good example is hydrogen combustion in a PWR containment. Hydrogen generated during core degradation can be released to the containment over several time periods. However, an important contribution to the uncertainty in containment loads generated by a combustion event is the total mass of hydrogen involved in a combustion event. One possibility is that hydrogen released to the containment over the entire in-vessel core damage period accumulates without being burned, perhaps as a result of the absence of a sufficiently strong ignition source. Molten core debris released to the reactor cavity at vessel breach could represent a strong ignition source, which would initiate a large burn (assuming the cavity atmosphere is not steam inerted). Because of the mass of hydrogen involved, this combustion event might challenge containment integrity. Another possibility is that while the same total amount of hydrogen is being released to the containment during in-vessel core degradation, a sufficiently strong ignition source exists to cause several small burns to occur prior to vessel breach. In this case, the mass of hydrogen remaining in the containment atmosphere at vessel breach would be very small in comparison to the first case, and the likelihood of a significant challenge to containment integrity at that time would be correspondingly lower. Therefore, the logic for evaluating the probability of containment failure associated with a large combustion event occurring at the time of vessel breach should be able to distinguish these two cases and preclude the possibility of a large combustion event if hydrogen was consumed during an earlier time frame.

Recognition of the interdependencies of phenomena. Most severe accident phenomena and associated events require certain initial or boundary conditions to be relevant. For example, a steam explosion can only occur if molten core debris comes in contact with a pool of water. Therefore, it may not be meaningful to consider ex-vessel steam explosions during accident scenarios in which the drywell floor (BWR) or reactor cavity (PWR) is dry at the time of vessel breach. Logic models for evaluating containment performance should capture these and many other such interdependencies among severe accident events and phenomena. Explicit representation of these interdependencies provides the mechanism for allowing complete traceability between a particular accident sequence (or PDS) and a specific containment failure mode.

3.5.2. Presentation of results

The total number of individual severe accident progressions represented by the Level 2 logic model can be quite large. Consequently, binning or grouping logic is often applied to determine the aggregate frequency of accident progressions that have common features. These features might include, time and/or mode of containment failure, manual actions to terminate core damage, or engineered safeguard system operation. As described in Section 3.7, if these features are selected appropriately, accident progressions can be grouped in a manner that allows common fission product source term to be assigned to them.

Regardless of how this grouping process is performed, the final results will have to be reviewed to ensure they are consistent with accident progression calculations performed for key accident sequences. Major contributors to various modes of containment failure will be identified and described. Results will be presented both in terms of total frequency of various levels of containment performance, and in terms of conditional probability, given core damage. Unusually high, or low, probabilities of containment survival (i.e. no failure) as well as important containment failure modes will be traceable to deterministic analysis of key accident progressions.

3.6. QUANTIFICATION OF CONTAINMENT EVENT TREE EVENTS

The review of a Level 2 PSA needs to include a detailed examination of the methods and technical bases used to define values for individual event probabilities in the probabilistic logic model. The methods used should be examined to ensure that calculated results from the PSA can be used to achieve the stated objectives of the study. The technical bases used to quantify events should be carefully examined to ensure they are traceable, and that the probabilities generated from them represent an unbiased characterization of accident behaviour. That is, appropriate consideration has to be given to the uncertainties that accompany deterministic calculations of severe accident phenomena.

3.6.1. Assignment of event probabilities

There are many approaches to transforming the technical evidence concerning containment loads and performance limits to an estimate of failure probability, but the following approaches appear most often in contemporary studies.

(1) In the first (least rigorous) approach, expert judgement is applied in translating qualitative terms expressing various degrees of uncertainty into quantitative (point estimate) probabilities. For example, terms such as ‘likely’ or ‘unlikely’ are assigned numerical values (such as 0.9 and 0.1). The subjectivity associated with this method is controlled to some extent by developing rigorous attributes for the amount and quality of information necessary to justify progressively higher confidence levels (i.e. probabilities approaching 1.0 or 0.0). The main concern about this method is that the estimates made may not be reproducible and may not provide a clear basis for understanding and resolving disagreements between a reviewer and the PSA team.

(2) The second technique involves a convolution of two probability density functions. In this technique, probability density functions are developed to represent the distribution of credible values for a parameter of interest (e.g. containment pressure load) and for its corresponding failure criterion (e.g. ultimate pressure capacity). The basis for developing these distributions is the collective set of information generated from plant specific integral code calculations, corresponding sensitivity calculations, other relevant mechanistic calculations, experimental observations, and expert judgement. The conditional probability of containment failure (for a given accident sequence) is then calculated as the convolution of the two density functions. It is important for the reviewer to realize that although an approach of this type may lead to a more traceable relationship between the estimated probability and the amount and quality of supporting data (e.g. code calculations & verification, experimental data), it is quite possible for an analyst to use unsupported judgements in developing the input probability distributions (i.e. simply ‘invent’ the distributions). Clearly, the reviewer should check this point.

(3) Decomposition methods are a more general form of the load-resistance comparison method described in (2). The basic idea is to break down a question such as ‘does the containment fail due to hydrogen combustion?’ into a set of questions that can be more easily analysed. For example, the question mentioned in the previous sentence might be broken down into a) ‘how much hydrogen is generated?’ b) ‘what is the hydrogen burn pressure, given x/y/z % hydrogen in the atmosphere, c) ‘what is the probability that the containment fails given a pressure rise of a/b?’ Such decompositions are often developed in the form of event trees. The problems a reviewer may encounter are similar to those that may be seen with method (2). While the questions addressed in the decomposition are chosen because they can be more easily related to information from experiments or code calculations, as with method (2), the reviewer may still encounter probabilities which have been assigned without adequate support. The physical reasonableness of the decomposition itself is also to be reviewed.

Most contemporary Level 2 PSAs use a mixture of approaches. It is particularly important for the reviewer to identify the method used to quantify events that are found to be important contributors to risk measures such as the frequency of early containment failure, or the frequency of large fission product releases. A meaningful interpretation of results takes into account situations where results may be heavily influenced by subjective values for the probability of 'unlikely' events.

3.6.2. Technical basis for event quantification

The input to the probabilistic models usually stems from several sources. For example, useful information should be available from:

- computer code calculations of severe accident behaviour,
- interpolation of results from code calculations,
- applications of relevant experiments,
- engineering calculations,
- expert judgement (possibly using all of the above sources), and
- engineered system and human reliability analysis.

The specific information used to support the assignment of event tree branch probabilities needs to be reviewed and compared to the following general guidelines. A quality Level 2 PSA will make maximum use of plant specific deterministic calculations. Use of generic information (e.g. reference plant analysis) needs to be justified, and is probably most appropriate for complex issues that are not treated by general purpose accident analysis codes (e.g. re-criticality following re-flood of a damaged reactor core, and steam explosions). Interpolation or extrapolation of results from code calculations is carefully examined to ensure that results are applied in a manner that is consistent with the framework of the original calculations. Use of 'reference plant' analysis is used only when accompanied by analysis or arguments that support its applicability to the plant under consideration. The reviewer has to explore any non-standard codes or hand calculations that were used with particular emphasis on assumptions.

Information derived from the containment system analysis (system unavailabilities, non-recovery probabilities, human error probabilities), should be reviewed with special attention paid to modelling consistency with relevant Level 1 PSA models.

3.6.3. Uncertainties in event quantification

The basic probability density functions representing uncertainty in each parameter involved in the containment performance logic model may be propagated throughout the entire model to allow for calculation of statistical attributes such as importance measures, and to allow for the generation of uncertainty distributions on results such as the frequency of source term bins.

One means of performing this propagation of uncertainties is the application of Monte Carlo sampling techniques (such as Latin Hypercube Sampling). The application of this technique to Level 2 PSA logic models, pioneered in Ref. [33], accommodates a large number of uncertain variables. Other techniques have been developed for specialized applications, such as the direct propagation of uncertainty technique developed to assess the probability of containment failure as a result of direct containment heating in a large dry PWR (see Ref. [36]). However, these other techniques are constrained to a small number of variables and are not currently practical for applications involving the potentially large number of uncertain variables addressed in a quality Level 2 PSA.

If an uncertainty analysis of the type described above has been performed, the reviewer needs to confirm that the probability distributions developed for key events reflect the full range of information on the subject.

However, in many Level 2 PSAs, comprehensive uncertainty analyses are not performed. In such cases, the reviewer needs to confirm that, at a minimum, sensitivity studies were performed to determine the extent to which Level 2 PSA results (e.g. frequency of various modes of containment failure) are influenced by the specific value of probability assigned to events in the CET model.

The review will also determine whether event quantification is influenced by a bias in the information used to evaluate severe accident phenomena. For example, the exclusive use of calculations performed with a single computer code can lead an analyst to high levels of confidence that a particular event is ‘certain’, or conversely ‘impossible.’ However, these conclusions may conflict with information developed in other studies, using a different computer code, for very similar circumstances. As mentioned in Section 2.2.4, it is necessary that the review team include experts in severe accident phenomena to ascertain whether such biases exist in the CET structure or in event quantification. In addition, PSA codes may have limitations which influence propagation of uncertainties and robustness of uncertainty analysis, such as limited capacity for event tree analysis, which forces the use of decomposition event trees. In these cases, the reviewers also have to carefully evaluate any uncertainty analysis performed for the PSA.

3.7. SOURCE TERMS CHARACTERIZATION

Estimates of fission product release to the environment (i.e. source terms) are typically, but not always, generated as a major product of a Level 2 PSA. If the scope of the Level 2 PSA is limited to an assessment of containment performance, source term analysis may not be necessary. Conversely, if the frequency of adverse public health and economic consequences are to be examined, a detailed source term characterization is essential. Therefore, the review of this element of a Level 2 PSA must be tailored to meet the objectives of the study. If the PSA objectives demand the characterization of fission product source terms, the assessment is plant specific; generic or qualitative source terms are generally not acceptable.

3.7.1. Source term binning process

As described in Section 3.5.2, results of the probabilistic analysis of containment performance are usually grouped according to major characteristics of severe accident progression. If a unique source term is assigned to each end state of the probabilistic logic model, these grouping characteristics include parameters that influence fission product evolution, retention and transport through each of the major barriers to the environment. End states grouped in such a manner are referred to as release categories or source term bins.

In some cases (e.g. containment bypass sequences) the definition of PDSs defines in itself the characteristics of the source term, and a binning process is not necessary.

The attributes used to define source term bins have to be reviewed to determine if accident progressions that are grouped into a common source term bin would, in fact, have similar radiological release characteristics and potential off-site consequences. These attributes are often plant and containment specific, but typical characteristics (for PWRs) are listed in Table IV. If a Level 3 PSA is to be performed using the results of the Level 2 source terms, additional attributes may be defined, such as location of release, energy of release, and release duration.

Verifying the similarity of source terms for accident sequences within a release category can be difficult without deterministic calculations of fission product release and transport. It is common practice to perform a source term calculation only for a single 'representative' accident progression within each release category. The reviewer needs to examine the accident progressions selected for representative source term calculations, and agree with the rationale used by the PSA analysts that other accident progressions within the same release category would result in a similar source term. The availability of calculations for alternative representative sequences in the most important source term categories would increase the reviewer's confidence in the results obtained.

3.7.2. Grouping of fission products

Fission products with common chemical and physical properties are usually treated collectively in severe accident source term analysis (see Ref. [6]). Distinctions among individual isotopes of major radionuclide species are not made in the calculation of fission product release to the environment. The grouping scheme is typically imbedded in the computer code used to generate source term estimates; a typical radionuclide grouping scheme is given in Table V.

Depending on the objectives and scope of the Level 2 PSA, a detailed accounting of all species of fission products may not be necessary. Occasionally, source term estimates are limited to the noble gases, I and Cs groups. This practice is generally acceptable because iodine and caesium release estimates tend to dominate the early and latent human health consequences, respectively. The reviewer needs to examine the method used to calculate radionuclide release to the environment, and be confident that the radionuclide grouping scheme is consistent with current, state of the art practices.

TABLE V. SOURCE TERM BINNING ATTRIBUTES

Release attribute	Possible attribute type
Time of release	Very early (containment failure prior to core damage or during core melt) Early (around the time of vessel breach) Intermediate (up to several hours after vessel breach) Late (to the end of Level 2 mission time)
Containment status at the end of Level 2 mission time	Containment by passed by interfacing systems LOCA Containment by-passed by unisolated steam generator tube ruptures (PWRs) Containment not isolated Containment penetration failure (enhanced leakage) Containment structural failure (large leak area) Containment vented (filtered/unfiltered) Basemat penetration Design basis leakage
Mode of ex-vessel releases	Dry core concrete interaction Core concrete interaction submerged No core concrete interactions
Fission product removal mechanisms	None Containment sprays and/or fan coolers operating (time of operation may be specified also) Secondary containment or reactor building
Pressure suppression pool (BWRs)	Sub-cooled Saturated By-passed (and time of bypass)
Time of core damage relative to accident initiation	Within a few hours After several hours (typically more than 10)

TABLE VI. RADIONUCLIDE CLASSES (MELCOR GROUPING)

Radionuclide Class Name	Representative Specie	Member Elements
Noble gases	Xe	He, Ne, Ar, Kr, Xe, Rn, H, N
Alkali metals	Cs	Li, Na, K, Rb, Cs, Fr, Cu
Alkaline earth	Ba	Be, Mg, Ca, Sr, Ba, Ra, Es, Fm
Halogens	I	F, Cl, Br, I, At
Chalcogens	Te	O, S, Se, Te, Po
Platinoids	Ru	Ru, Rh, Pd, Re, Os, Ir, Pt, Au, Ni
Early transition elements	Mo	V, Cr, Fe, Co, Mn, Nb, Mo, Tc, Ta, W
Tetravalents	Ce	Ti, Zr, Hf, Ce, Th, Pa, Np, Pu, C
Trivalents	La	Al , Sc, T, La, Ac, Pr, Nd, Pm, Sm Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf

3.7.3. Fission product release and transport calculations

The analytical models (computer codes) used to calculate fission product release and transport should be verified as being appropriate for the task. Common computer codes used for this purpose are listed in Table II.

If the objective of the Level 2 PSA is to provide a technical basis for installing (or not installing) severe accident mitigation devices, such as a filtered containment venting system, independent source term calculations, using a different computer code are strongly recommended. Adaptation of source terms from reference plants should not be accepted. Independent calculations of source terms for *selected* sequences may be warranted if the frequency of large radionuclide release (i.e. fractional releases of volatile species exceeds 10%) is unusually high, or if the PSA is to be extended to Level 3 analysis.

In many cases, however, source term results are used simply as a quantitative measure for ranking the relative importance of various accident sequences. Under such circumstances, a detailed review of calculated results may not be warranted. However, spot checks of results have to be made by comparison to those documented in other similar studies (see Ref. [33] and Ref. [37]).

If the frequency of the following accident conditions is significant, the corresponding source terms need to be reviewed with particular care:

- Steam generator tube ruptures; releases from unisolated steam generator tube ruptures can span a very broad range. Very large releases can accompany accident sequences in which the steam generator secondary inventory is depleted; conversely, moderate release may result if the ruptured tube(s) is submerged.
- Releases from accidents with unisolated containment; depending on the size of the failed isolation(s), and on the path of release, estimates may vary from small to very large.
- Releases from accidents with late containment failure; depending on the containment capacity, late failure may occur anywhere between 10 hours and 48 hours after core damage. Over these long time period, revaporization of volatile species I, Cs, and Te from dry, overheated surfaces can dominate the source term.

- Releases from accidents with scrubbing provided by containment sprays; the effectiveness of containment spray in reducing airborne radionuclide concentrations can span several orders of magnitude, depending on spray water temperature, droplet size and spray distribution within the containment atmosphere.

3.7.4. Treatment of uncertainties in source terms estimates

Quantitative evaluations of source term uncertainties are not usually made in Level 2 PSAs. However, it is necessary that a structured sensitivity analysis of source term calculations for major accident scenarios be available and reviewed. The reviewer can then verify that major modelling assumptions are identified and their importance quantified. For example, the extent to which iodine is assumed to be permanently retained in water pools (e.g. BWR suppression pools) during late phases of an accident is highly uncertain. The effects of baseline modelling assumptions concerning iodine aqueous chemistry (and many other similar processes) need to be measured and incorporated in the Level 2 PSA results.

3.7.5. Presentation of results

Presentation of source terms results has to conform to the prescriptions detailed in Ref. [6]. In case the reviewers have developed models for an independent estimate of source terms, similar tables need to be derived and a comparison made with the results of the PSA. In addition, from these tables, cumulative complementary distribution functions may be constructed, and compared with results of published Level 2 PSAs. This information is vital for the review process, and can provide insights on several risk figures of merit (including large early release frequency — LERF).

3.8. RESULTS OF THE LEVEL 2 PSA

The general statements made in Ref. [5] about how the results of a Level 1 PSA should be presented are also valid for a Level 2 PSA. This is particularly important for the Level 2 PSA in view of the complicated phenomena modelled and the uncertainties involved. Difficulties inevitably arise in the communication of the results of the analysis to non-specialists.

This places a more onerous requirement on analysts and reviewers to present the results of the PSA and the findings clearly and succinctly, in non-specialist language, so that they can be understood more widely. This is particularly important where the results have been used to indicate that changes need to be made to the design or operation of the plant to provide additional protection for severe accidents.

3.8.1. Review of the results of the PSA

The presentation of the results of the Level 2 PSA would depend on what the aims and objectives were of the analysis. For a full scope analysis, the results would be in the form of source terms and their frequencies where the source term would specify the quantity of each of the isotopes for each release groups included in the analysis. This could be in a summary table as shown in Table XX of Ref. [6]. This information needs to be grouped to provide estimates of the frequency of a large release or an large early release where this is required to allow a comparison to be made with probabilistic safety criteria.

The results have to include sufficient information to give insights into the main contributors to the risk and the uncertainties in these estimates of the risk. This would identify weaknesses in the design or operation of the plant in providing protection against severe accidents.

The reviewers have to be satisfied that the global results of the PSA are plausible, the interpretation and conclusions drawn from the results are logical and correct, and the overall objectives of the PSA and the PSA requirements and guidelines are met.

The reviewers have to check that a sufficient range of sensitivity studies have been carried out which relate to the aspects of the analysis which are most significant in determining the level of risk and those which have the highest uncertainty. The reviewers need to check that the results of the sensitivity studies demonstrate that the conclusion of the analysis and the insights derived from it are still valid.

It is suggested that the results of the Level 2 PSAs be compared with those for plants with similar containment and containment systems design and any differences identified. These should be investigated since this may provide additional help to the reviewers in the identification of potential weaknesses of the PSA.

The reviewers need to check the assumptions made in the PSA carefully. This applies particularly to areas of the PSA that rely on expert judgement. The reviewers should identify relevant experimental data, which address processes represented in analytical models contained in the PSA, and satisfy themselves that this has been properly and adequately taken into account.

The reviewers should be satisfied that the benefits from carrying out accident management measures are reasonable in relation to the results of the PSA.

3.8.2. Use of the results of the PSA

The results of the analysis are to be compared with the probabilistic safety criteria defined for the plant (if such goals have been defined). In some countries, risk criteria have been defined which relate to the frequency of a large release or a large early release of radioactivity.

The results of the PSA are to be used to determine whether there are any weaknesses in the design and operation of the plant. Where such weaknesses are identified, consideration may be given to identifying improvements which could be made to reduce the risk from severe accidents. This typically includes additional safety systems to provide protection for some of the adverse consequences of a severe accident. In the past, such additional safety systems have included the following:

- the incorporation of hydrogen igniters or recombiners which have sufficient capacity to deal with the rate of hydrogen generation which would occur during a severe accident.
- the addition of a filtered containment venting system which would prevent failure of the containment in a longer time due to overpressurization.

The results of the PSA may also be used to determine whether there are additional accident management measures which could be incorporated to reduce the risk from severe accidents. This typically includes the use of existing equipment to provide protection for some of the adverse consequences of a severe accident. In the past, such accident management measures have included the following:

- the use of the primary relief valves to depressurize the primary circuit to prevent the possibility of high pressure melt ejection.
- the addition of water to the containment to help with core cooling.

The reviewers need to check that, where such accident management measure have been identified which are effective in reducing the risk, they have been included explicitly in the emergency operating instructions.

3.9. AUDIT OF THE PSA QUALITY ASSURANCE

As discussed in Sections 2.2 to 2.4, it is good practice for the QA procedures used in performing the PSA (including technical procedures) to be reviewed and approved by the regulatory authority at an early stage of a PSA (ideally, before actual analysis starts). Whether or not this is done, the regulatory authority may conduct audits during the process of the PSA development to ensure that the QA procedures are indeed followed, and that the process for performing PSA is being properly managed. The frequency of an audit can be determined to meet specific needs. To receive the maximum benefit from the audits, it is recommended to conduct the first one at an early stage in the PSA development, so that any deficiencies identified in the audit can be corrected then.

ABBREVIATIONS

ADS	automatic depressurization system
APET	accident progression event tree
ATWS	anticipated transient without scram
BWR	boiling water reactor
CDF	core damage frequency
CET	containment event tree
CNRA	committee of Nuclear Regulatory Activities
CSNI	committee of the Safety of Nuclear Installations
CT	crisis team
DET	decomposition event tree
ECCS	emergency core cooling system
EOP	emergency operation procedures
FCI	fuel coolant interactions
HEP	human Error Probability
HRA	human reliability analysis
IPERS	International Peer Review Service
ISPART	International Probabilistic Safety Assessment Review Team
LERF	large early release frequency
LOCA	loss of coolant accident
PDS	plant damage state
PORV	power operated relief valve
PSA	probabilistic safety assessment
PWR	pressurized water reactor
QA	quality assurance
RCS	reactor coolant system
RPV	reactor pressure vessel
SGTR	steam generator tube rupture

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