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# IAEA SAFETY STANDARDS

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# Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants

**DRAFT SPECIFIC SAFETY GUIDE**

**DS528**

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# 1. INTRODUCTION

## BACKGROUND

1.1. IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [1], establishes principles to ensure the protection of workers, the public and the environment, now and in the future, from harmful effects of ionizing radiation. These principles emphasize the need to assess and manage the risk posed by nuclear facilities. In particular, para. 3.22 of SF-1 [1] on optimization of protection states:

> "To determine whether radiation risks are as low as reasonably achievable, all such risks, whether arising from normal operations or from abnormal or accident conditions, must be assessed (using a graded approach) a priori and periodically reassessed throughout the lifetime of facilities and activities."

1.2. Several IAEA Safety Requirements publications establish general and specific requirements on risk assessment for nuclear power plants. Paragraph 4.13 of IAEA Safety Standards No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2]) states:

> "The safety assessment shall include a safety analysis, which consists of a set of different quantitative analyses for evaluating and assessing challenges to safety by means of deterministic and also probabilistic methods."

1.3. Paragraph 4.55 of GSR Part 4 (Rev.1) [2] further states:

> "The objectives of a probabilistic safety analysis are to determine all significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined."

1.4. Requirement 42 of IAEA Safety Standards No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3], states:

> "A safety analysis of the design for the nuclear power plant shall be conducted in which methods of both deterministic analysis and probabilistic analysis shall be applied to enable the challenges to safety in the various categories of plant states to be evaluated and assessed."

1.5. Paragraph 5.76 of SSR-2/1 (Rev. 1) [3] further states (footnote omitted):

> "The design shall take due account of the probabilistic safety analysis of the plant for all modes of operation and for all plant states, including shutdown, with particular reference to:
>
> (a) Establishing that a balanced design has been achieved such that no particular feature or postulated initiating event makes a disproportionately large or significantly uncertain contribution to the overall risks, and that, to the extent practicable, the levels of defence in depth are independent;

(b) Providing assurance that situations in which small deviations in plant parameters could give rise to large variations in plant conditions (cliff edge effects) will be prevented;

(c) Comparing the results of the analysis with the acceptance criteria for risk where these have been specified."

Thus, a full scope probabilistic safety assessment (PSA) will contribute to assess and verify the safety of nuclear power plants in relation to potential internal initiating events and internal and external hazards as well as their combinations.

1.6. PSA has been shown to provide important safety insights in addition to those provided by deterministic analysis. PSA provides a methodological approach for identifying accident sequences that can follow from a broad range of initiating events and it includes a systematic and realistic determination of accident frequencies and consequences. In international practice, three sequential levels of PSA are generally recognized:

(1) In Level 1 PSA, the design and operation of the plant are analysed in order to identify the sequences of events that can lead to core and/or fuel damage and the corresponding core and/or fuel damage frequencies are estimated. Level 1 PSA provides insights into the strengths and weaknesses of structures, systems and components (SSCs) important to safety and procedures in place or envisaged as preventing core and/or fuel damage. Further information is provided in IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [4].
(2) In Level 2 PSA, the chronological progression of core and/or fuel damage sequences identified in Level 1 PSA is evaluated, including a quantitative assessment of phenomena arising from severe damage to reactor fuel and/or to spent fuel. Level 2 PSA identifies ways in which associated releases of radioactive material from fuel can result in releases to the environment. It also estimates the frequency and other relevant characteristics of releases of radionuclides to the environment. This analysis provides additional insights into the relative importance of accident prevention and mitigation measures and the physical barriers to the release of radioactive material to the environment (e.g. a containment building).
(3) In Level 3 PSA, public health and other societal consequences are estimated, such as the contamination of land or food from the accident sequences that lead to a release of radioactive material to the environment.

1.7. PSAs are also classified according to the range of initiating events (internal and/or external to the plant) and plant operating modes that are to be considered.

1.8. Level 2 PSA is a structured process. Although there may be differences in the approaches for performing a Level 2 PSA, the general main steps are shown in FIG. 1 and are as follows:

(a) Level 1 PSA provides information on the accident sequences that lead to fuel damage and hence provides the starting point for Level 2 PSA. The accident sequences identified by Level 1 PSA may not include information on the status of the SSCs dedicated to ensuring the confinement function (e.g. the containment systems in pressurized water reactors) that mitigate the effects of severe accidents.
(b) The interface between Level 1 PSA and Level 2 PSA is where the accident sequences leading to fuel damage are grouped into plant damage states (PDSs) based on similarities

in the plant conditions that determine the further accident progression. If the status of SSCs dedicated to ensuring the confinement function was not addressed in the Level 1 PSA, it needs to be considered by means of so-called 'bridge trees' of the interface between Level 1 PSA and Level 2 PSA or by extended Level 1 event trees, as the first step of the Level 2 PSA.

(c) An accident progression event tree[1] is used to model accident progression to identify accident sequences that challenge the SSCs dedicated to ensuring the confinement function and lead to releases of radioactive material to the environment.

(d) Source term analysis is used to determine the quantities and timings of radioactive material released to the environment from each of the release categories.

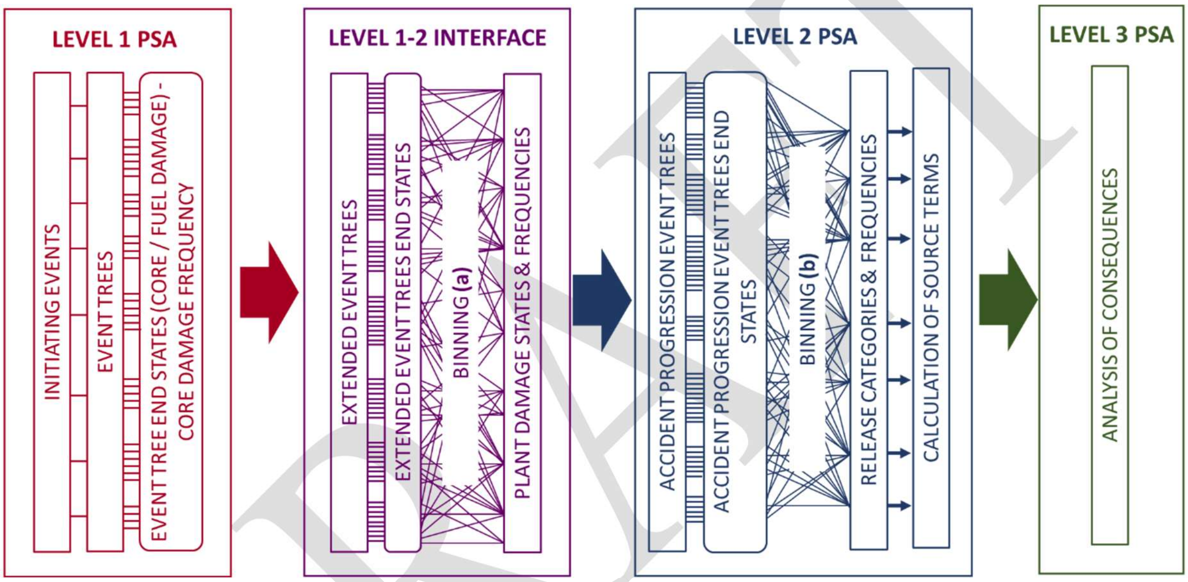


*FIG. 1. General overview of the development of a typical Level 2 PSA.*

1.9. The process for carrying out Level 2 PSA is not unique, but rather depends on the approach selected. For practical purposes, a number of grouping tasks sometimes need to be performed for Level 2 PSA at the following stages, as indicated in FIG. 1.

(a) The grouping (binning) of the core and/or fuel damage sequences (extended to include the status of SSCs dedicated to ensuring the confinement function) into the PDSs that form the starting point for the Level 2 PSA. Some methodologies us a multi-step process of grouping and re-grouping similar PDSs into a condensed set of PDSs to be taken forward into the accident progression event tree analysis.

(b) The grouping (binning) of the severe accident sequences identified in the accident progression event tree analysis into release categories.



[1] Such event trees are also termed containment event trees. The term accident progression event trees has been chosen throughout this safety guide, like in the ASAMPSA2 project [21], because it is more generally applicable.

1.10. Further grouping or regrouping of the release categories into a condensed set[2] that would be taken forward into the Level 3 PSA may be needed . The interface between Level 2 PSA and Level 3 PSA is not addressed in detail in this document although it is touched upon in section 15 on the use and applications of Level 2 PSA.

1.11. Level 1 PSA and Level 2 PSA of varying scope and level of detail have been performed for almost all nuclear power plants worldwide in operation or under construction, whereas Level 3 PSA has been performed for some nuclear power plants in some Member States.

1.12. This Safety Guide was prepared on the basis of a systematic review of relevant IAEA publications, including Refs [1][6] and an International Nuclear Safety Group (INSAG) report [7].

1.13. This Safety Guide replaces IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants[3], which it supersedes.

OBJECTIVE

1.14. The objective of this Safety Guide is to provide recommendations for meeting the requirements of GSR Part 4 (Rev.1) [2] in performing or managing a Level 2 PSA project for a nuclear power plant; this Safety Guide therefore complements IAEA Safety Standards No. SSG-3 (Rev. 1), Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [4]. One of the aims is to promote a standard framework, standard terms and a standard set of documents for PSAs to facilitate regulatory and external peer review particularly for Level 2 PSA results.

1.15. This Safety Guide also provides a consistent, reliable means of ensuring the effective fulfilment of obligations under Article 14 of the Convention on Nuclear Safety [8].

1.16. The recommendations presented in this Safety Guide are based on internationally recognized good practices for current water cooled reactors. However, they are not intended to pre-empt the use of equivalent new or alternative methods. On the contrary, the use of any method that achieves the objectives of Level 2 PSA is acceptable. Although the recommendations provided in this Safety Guide are intended to reflect a technology inclusive methodology, the details of the analysis methods might change as understanding of severe accident phenomena improves or to adapt to a particular reactor technology. Most of the phenomenology described as examples in this Safety Guide is directly applicable to current water cooled reactors, such as pressurized water reactors or boiling water reactors but may also



[2] Some methodologies use the term 'source term categories' to denote the final condensed set of release categories used for source term calculations. It should be noted however that both terms 'release category' and 'source term category' are generally used synonymously to mean a group of accident progression sequences that would generate a similar source term to the environment. The categories are defined by attributes in relation to the release. See also footnote 15.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4, IAEA, Vienna (2010).

apply or adapt to a particular reactor technology. For example, for Molten Salt Reactors with liquid fuel, the concept of core melt is not meaningful.

SCOPE

1.17. This Safety Guide addresses the necessary methodological technical features of Level 2 PSA for nuclear power plants (both existing and new plants) in relation to its application, with an emphasis on the procedural steps and essential elements of the PSA rather than on details of the modelling methods. This Safety Guide includes all the steps in the Level 2 PSA process, up to and including the determination of the detailed source terms needed as input into a Level 3 PSA.

1.18. The scope of a Level 2 PSA addressed in this Safety Guide includes all modes of normal operation of the plant (i.e. startup, power operation, shutting down, shutdown, maintenance, testing and refuelling) and considers the Level 1 PSA results obtained for all potential initiating events and potential hazards, (i.e. a full scope Level 1 PSA as described in SSG-3 (Rev. 1) [4]), namely: (a) internal initiating events caused by random component failures and human error, (b) internal hazards and (c) external hazards, both natural and human induced, as well as combinations of hazards, such as consequential (subsequent) events, correlated events and unrelated (independent). If the objectives of the Level 2 PSA are limited, only the relevant recommendations provided in this Safety Guide apply; if the scope of the Level 1 PSA is limited (see paras 2.8-2.9), additional analysis to that described in this Safety Guide may need to be carried out.

1.19. If the aim of the PSA is to determine all the contributions to risk to public health and society, then the PSA will need to take into account in the calculation of the source term the potential for release from other sources of radioactivity from the plant, such as irradiated fuel and stored radioactive waste. Such an aim is not detailed in this Safety Guide, which focuses on releases of radioactive material resulting from severe accidents in the reactor and the spent fuel pool. This Safety Guide also covers the development of Level 2 PSA for sites where several units are located, which may be considered given that national regulatory requirements compel such studies, as part of the quantification of the source term at the site level.

1.20. Different plant designs have different provisions to prevent or limit the release of radioactive material following a severe accident. Most designs include a containment structure or building (hereinafter referred to as 'containment') as one of the passive features dedicated to ensuring the confinement function. The phenomena associated with severe accidents are also very much influenced by the reactor technology, design and composition of the reactor core. The recommendations provided in this Safety Guide are intended to be technology inclusive to the extent possible. However, the number and content of the various steps of the analysis assume the existence of some type of containment with related passive features, and the associated phenomena with regard to the nuclear reactor technology.

1.21. Recommendations relating to the performance, project management, documentation and peer review of a PSA and implementation of a management system in accordance with IAEA Safety Standards No. GSR Part 2, Leadership and Management for Safety [5] are provided in SSG-3 (Rev. 1) [4] and are therefore not addressed here. This Safety Guide addresses only the aspects of PSA that are specific to Level 2 PSA.

STRUCTURE

1.22. Sections 2–12 of this Safety Guide provide recommendations on the performance of Level 2 PSA, with each section corresponding to a major procedural step in Level 2 PSA as shown in Fig. 2. Section 13 provides recommendations on the performance of Level 2 PSA for the spent fuel pool. Section 14 provides recommendations on the performance of Level 2 PSA for a site with multiple nuclear power plants (also known as a multi-unit site). Section 15 provides recommendations on the uses and applications of a Level 2 PSA. The Appendix gives an overview of human reliability analysis in Level 2 PSA. Annex I discusses various types of computer code available for simulation of severe accidents and PSA studies. Annex II presents a sample outline of documentation for a Level 2 PSA. Annex III provides information on the common risk metrics used in Level 2 PSA with examples from several Member States.

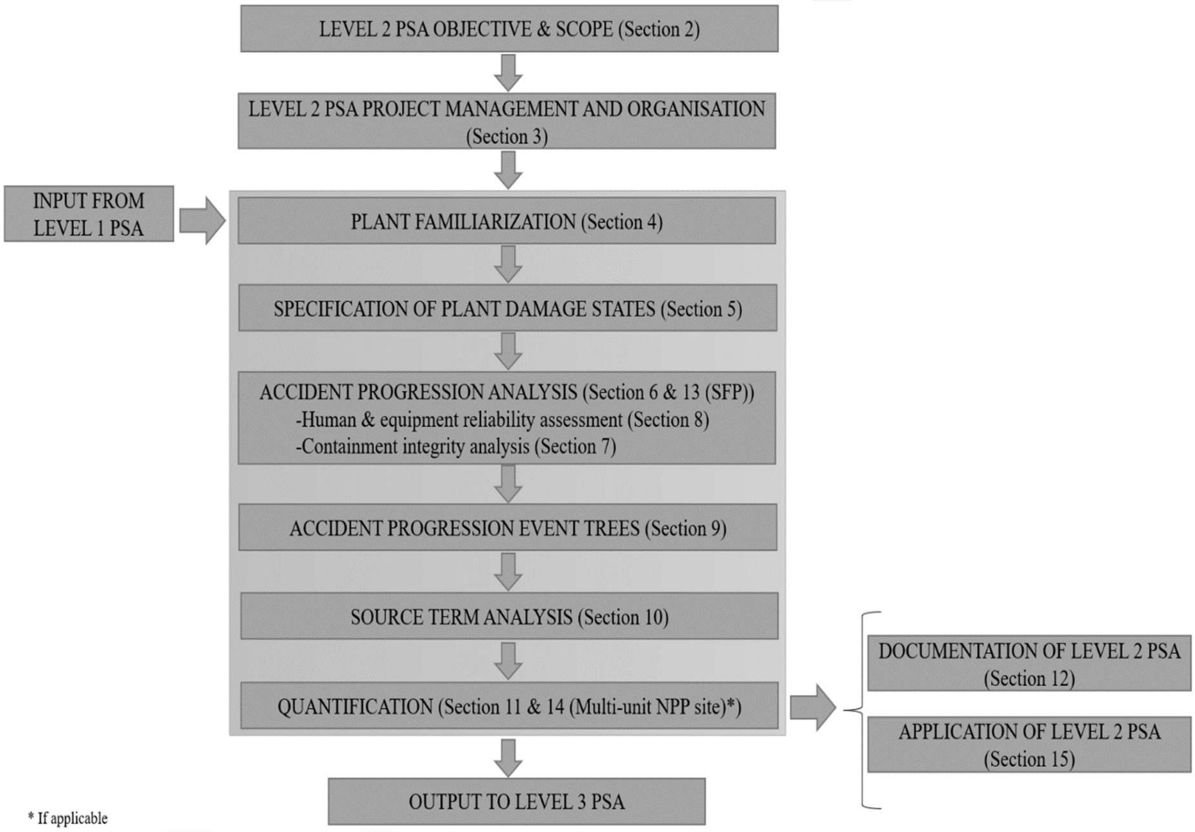


*FIG. 2. Main steps in the performance of Level 2 PSA.*

## 2. GENERAL CONSIDERATIONS RELATING TO THE PERFORMANCE AND USE OF LEVEL 2 PSA

### OBJECTIVES OF LEVEL 2 PSA

2.1. Requirement 4 of GSR Part 4 (Rev. 1) [2]states:

**“The primary purposes of the safety assessment shall be to determine whether an adequate level of safety has been achieved for a facility or activity and whether the basic safety objectives and safety criteria established by the designer, the operating organization and the regulatory body…have been fulfilled.”**

2.2. The main objective of Level 2 PSA is to determine if sufficient safety provisions[4] have been made to manage a severe accident and to mitigate the effects of such an accident to ensure that sufficient protection of the population and the environment has been achieved and, for new reactor designs, contribute to demonstrating the ‘practical elimination’ [5] of plant event sequences that could lead to an early radioactive release or a large radioactive release (see also IAEA Safety Standards Series No. SSG-88, Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants [9]. Another objective of Level 2 PSA is to demonstrate the balanced design of safety provisions to manage severe accidents and mitigate their effects. The sufficiency and the balanced design of safety provisions to manage severe accidents and mitigate their effects are normally demonstrated respectively by compliance with numerical safety goals (see paras 2.16–2.17) and by analysis of their individual contributions to the overall risk profile. The safety provisions to manage severe accidents and mitigate their effects could include:

(a) Systems provided specifically to mitigate the effects of the severe accident, such as in-vessel or ex-vessel molten core retention features, hydrogen mixing devices or hydrogen recombiners, or filtered containment venting systems;
(b) The inherent strength of the containment or the capability for confinement and retention of radioactive material within dedicated SSCs, and the use for accident management of equipment provided for other purposes;
(c) Guidance to plant operators on severe accident management.

2.3. The objectives of Level 2 PSA should be defined. These can include the following:

(a) To gain insights into the progression of severe accidents and the performance of the confinement function, ensured by dedicated SSCs (e.g. the containment), to minimize the release of radioactive material;
(b) To identify plant specific challenges and vulnerabilities of the dedicated SSCs ensuring the confinement function with regard to severe accidents;
(c) To provide an input into the resolution of specific regulatory concerns;
(d) To provide an input into determining compliance with the probabilistic safety goals, or with probabilistic safety criteria if these have been set. Most common probabilistic safety goals or criteria relate to large release frequencies and/or large early release frequencies, as further explained in para 2.17;

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[4] ‘Safety provisions’ are considered in this Safety Guide as design solutions applied to SSCs and related operational strategies.
[5] The recommendations for new nuclear power plants on the implementation of selected requirements in SSR-2/1 (Rev. 1) that are related to defence in depth and practical elimination of event sequences leading to early radioactive releases or large radioactive releases are presented in a Specific Safety Guide and are not considered in this Safety Guide.

(e) To identify the major failure modes of dedicated SSCs ensuring the confinement function (e.g. containment failure modes) and their frequencies, and to estimate the associated frequencies and magnitudes of radionuclide releases;
(f) To provide an input into the development of off-site emergency preparedness and response arrangements;
(g) To provide an input into the development of plant specific accident management guidance and strategies;
(h) To provide an input into determining plant specific options with regard to design and accident management guidelines and strategies aiming to risk reduction;
(i) To provide an input into the prioritization of research activities for the minimization of risk significant uncertainties;
(j) To provide an input into Level 3 PSA consistent with the PSA objectives;
(k) To provide an input into the environmental impact assessment of the plant;
(l) For new reactor designs, to contribute to demonstrating the 'practical elimination' of plant event sequences that could lead to an early radioactive release or a large radioactive release;
(m) To gain insights into possible cliff edge effects leading to radioactive releases.

2.4. Each of these objectives would place distinct emphasis on one of the various aspects of the Level 2 PSA. The objectives reflecting the intended uses and applications of the Level 2 PSA should therefore be clearly specified at the beginning of the Level 2 PSA project. In particular for the design stage, the detail of Level 2 PSA should be sufficient to achieve the above mentioned objectives considering the difficulty or impossibility to implement design safety features to manage severe accidents in a later stage.

SCOPE OF LEVEL 2 PSA

2.5. Requirement 1 of GSR Part 4 (Rev. 1) [2] states:

> **"A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out at a particular stage for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity."**

While Requirement 14 of GSR Part 4 (Rev. 1) [2] states that **"The performance of a facility or activity in all operational states and, as necessary, in the post-operational phase shall be assessed in the safety analysis."**

2.6. In undertaking a Level 2 PSA, there are two types of approaches likely to be encountered depending on the overall objective of the PSA project and the software capabilities for developing the probabilistic models. The first is an integrated approach where the Level 1 and Level 2 PSA models are developed, linked and quantified in a single software tool. The second is a separated approach, where the Level 1 and Level 2 PSA models are not developed, linked or quantified in a single software tool such that additional steps to transfer data / information / results from Level 1 to Level 2 would be required. ASAMPSA2 provides information on the advantages and disadvantages of each approach [21]. The integrated approach has mainly been applied to the latest Level 2 PSA developments for new nuclear power plants equipped with water cooled reactors, but also as an alternative for advanced nuclear power plant designs equipped with non-water cooled reactors for which significant core degradation[6] is not in the scope of the analysis. In the separated approach, the Level 2 PSA is performed after the Level 1 PSA is complete, when some additional system analyses may be necessary. If the Level 2 PSA is performed following an integrated approach, the requirements of the Level 2 PSA should be fed into the Level 1 PSA; in this way, all plant related features that are important to the analysis of the response of dedicated SSCs ensuring the confinement function and the analysis of the source terms will be considered wherever possible in the Level 1 PSA. In either approach, when linking the Level 1 and Level 2 PSA models, typically via the specification and quantification of PDSs, it should be ensured that the Level 2 PSA takes fully into account the initial and boundary conditions from the Level 1 PSA model and the dependencies between the Level 1 PSA and the Level 2 PSA.

2.7. The scope of Level 2 PSA should be determined by its defined objectives, see paras 2.2-2.3 and its specific intended uses and applications, as further detailed in para 15.2. Appropriate consideration should be given to the significance of key uncertainties associated with phenomena and the modelling as described in the following sections of this safety guide. Care should be taken to avoid distorting the conclusions of the Level 2 PSA through models and assumptions that are systematically biased towards particular outcomes (often for the sake of conservatism).

2.8. Commonly, the Level 2 PSA is developed as a base model for a comprehensive list of PDSs related to internal events. This base model should be used for the extension to relevant operating modes and to internal and external hazards (see Ref. [10]).

2.9. If the starting point of Level 2 PSA is an existing Level 1 PSA, then its output may not explicitly cover all the features that need to be taken into account. For example, if the objective of the Level 1 PSA was the quantification of core damage frequency, then the status of the dedicated SSCs ensuring the confinement function, such as the containment and the containment safety systems, may not have been directly addressed and therefore will have to be determined as part of the Level 2 PSA or as part of the modelling of the interface between Level 1 and Level 2 PSA (e.g. specification and quantification of the PDSs).

2.10. If the scope of the PSA includes internal and/or external hazards (e.g. fire, earthquake, human induced hazards (see SSG-3 (Rev. 1) [4] and Ref. [10])), their potential impact on the confinement function and the dependent failures they could cause should be taken into account as part of the Level 2 PSA, if they have not been previously taken into account in the Level 1 output. Examples of such dependent failures include failures in the containment isolation system due to cable fire, damage of containment structures due to seismic events or transportation accidents.

2.11. If the spent fuel pool is located inside the same containment housing the reactor (e.g. in some pressurized water reactor and boiling water reactor designs), the Level 2 PSA should consider the simultaneous consequences of severe accident phenomena induced by the reactor

[6] The notion of “significant core degradation” for some non-water cooled reactor technologies, which might not have a "reactor core” as it is conventionally understood for water cooled reactors, might not be applicable. However, the analysis would aim at identifying the challenges to the containment due to contact of the heat source (e.g. fuel salt) and related phenomena that might lead to radioactive releases.

core and the spent fuel pool for this containment and the source term calculations. Recommendations on Level 2 PSA for the spent fuel pool are provided in section 13 of this Safety Guide.

2.12. If the scope of the PSA includes sources of radioactivity (e.g. liquid radioactive waste or dry long-term storage of spent fuel) located outside of the containment (e.g. reactor containment building), then the potential risk of release from those sources should be considered. As stated in para 1.19, releases from other sources of radioactivity from the plant, such as irradiated fuel and stored radioactive waste, are not detailed in this safety guide.

2.13. Any analysis and assumptions associated with a Level 2 PSA should be as realistic as possible, commensurate with the intended uses and applications of the Level 2 PSA, and include an uncertainties assessment, consistent with the intent and scope of the study being undertaken.

2.14. If the scope of the PSA study considers the Level 3 PSA, the scope of the Level 2 PSA should consider the input requirements needed to conduct the Level 3 PSA.

2.15. If several reactor units (e.g. power and/or research reactors) are located at the site, the scope of the Level 2 PSA might include the impact of severe accidents for the accident management of more than one unit on the site and the corresponding aggregation of risk for these units on the site. Further recommendations on conducting a multi-unit Level 2 PSA are provided in section 14 of this Safety Guide.

## REFERENCE VALUES, PROBABILISTIC SAFETY GOALS OR CRITERIA AND RISK METRICS FOR LEVEL 2 PSA

2.16. The general recommendations related to reference values, probabilistic safety goals or criteria and risk metrics used in PSA presented in paras 2.10–2.15 of SSG-3 (Rev. 1) [4] are applicable to Level 2 PSA and are not repeated here. Paragraphs 2.17–2.19 provide recommendations on meeting Requirement 4 of GSR Part 4 (Rev. 1) [2] in relation to reference values and risk metrics for Level 2 PSA.

2.17. Level 2 PSA risk metrics should provide information that is meaningful and sufficient to facilitate the use and interpretation of the PSA with regard to the risk profile of the nuclear power plant. This could be represented by a sufficiently low frequency of occurrence of releases from the containment in exceedance of a magnitude of fission products. Often, a temporal element is also included such that effective public safety measures for sheltering or evacuation can be undertaken. Large release frequency and large early release frequency are the most common risk metrics used in Level 2 PSA, but there is variation among Member States (see Annex III). A large release means a release of radioactive material from the plant with significant off-site impacts necessitating off-site emergency arrangements. The release can be specified in a number of ways including the following:

(a) As absolute quantities (in becquerels) of the most significant radionuclides released;
(b) As a fraction of the inventory of the core;
(c) As a specified dose to the most exposed person off the site;
(d) As a release resulting in unacceptable consequences.

2.18. The definition of Level 2 PSA risk metrics should provide adequate information to understand the meaning of qualitative concepts or terms to be derived as quantitative values.

Therefore, in defining the Level 2 PSA risk metrics the terms “large”, “early”, “release”, “exceedance” and “frequency” should be considered, as follows:

(a) The defined limits related to unacceptable consequences in terms of dose for the general public and the impact in the environment regarding limits in space and time;
(b) The evaluation of capabilities associated with severe accident management programmes and with the emergency preparedness and response plan to effectively arrest and control severe accident progression and implement off-site emergency response actions;
(c) The design capabilities of SSCs to effectively retain and reduce the energy, the quantity and the physical form of the chemical elements and the radioisotopes contained in the fuel, the reactor core and the reactor coolant system;
(d) The radiotoxicity of chemical elements and the radioisotopes that could be liberated as result of the accident sequences;
(e) The expected measure in a time frame of the probability of occurrence of severe accidents;
(f) The uncertainties associated with the assumptions and the results of the Level 2 PSA study.

2.19. The following should also be taken into consideration in defining Level 2 PSA risk metrics:

(a) Current definition of probabilistic safety goals or criteria in use in other Member States;
(b) Operating experience feedback;
(c) The relationship between defined safety goals related to different PSA levels (e.g. between core damage frequency and large early release frequency);
(d) Implications of exceedance of probabilistic safety goals or criteria;
(e) Strategies to cope with the exceedance of probabilistic safety goals or criteria.

LIVING PSA

2.20. Requirement 24 of GSR Part 4 (Rev.1) [2] states that “**The safety assessment shall be periodically reviewed and updated**”, while Requirement 12 of SSR-2/2 (Rev.1) [11]states:

> “**Systematic safety assessments of the plant, in accordance with the regulatory requirements, shall be performed by the operating organization throughout the plant’s operating lifetime, with due account taken of operating experience and significant new safety related information from all relevant sources.**”

2.21. In the operating lifetime of a nuclear power plant, modifications are often made to the SSC design or to the way the plant is operated. Such modifications could have an impact on the level of risk associated with the plant which, in the context of this Safety Guide, is represented by the risk metrics associated with Level 2 PSA. Additional statistical data on the frequencies of initiating events, the probabilities of component failure and severe accident management guidelines will become available during plant operation. Likewise, new information and state-of-the-art methods, tools and research results related to severe accident may become available, which may change some of the assumptions made in the analysis and hence the risk estimates given by the Level 2 PSA. Consequently, the PSA should be kept up to date throughout the lifetime of the plant to ensure that it remains relevant to the decision making process. A PSA that undergoes periodical updating is termed a ‘living PSA’. The updating of a PSA should be initiated by a specified process, and the status of the PSA should be reviewed regularly to ensure that it is maintained as a representative model of the plant and is fit for purpose.

2.22. Data should be collected throughout the lifetime of the plant to check or update the PSA. These should include data on operating experience, in particular data on initiating events, data on component failures and unavailability during periods of testing, maintenance and repair, and data on human performance. The results from the PSA should be periodically reassessed in the light of new data.

2.23. The development of a living PSA should be encouraged, to assist the decision making process in the normal operation of the plant. Many issues, such as evaluation of the change in risk associated with a change to the plant or a temporary change in the allowed outage time of a component, can be supported by arguments derived from a PSA. Experience has shown that such a living PSA can be of substantial benefit to the operating organization and its use is generally welcomed by regulators (see Ref. [12]).

USE OF PSA IN THE DECISION MAKING PROCESS

2.24. This section describes some general issues relevant to the performance of PSA and the use of PSA results in practice. Although the scope of the Safety Guide is limited to consideration of Level 2 PSA, this section describes the issues from a broader perspective in order to provide a complete picture of the capabilities of PSA methodology and its results. Some statements in this section do not represent explicit recommendations; rather, they provide supporting information to facilitate understanding of the context of other statements and recommendations provided in other sections of the Safety Guide.

2.25. Paragraphs 2.25–2.26 provide recommendations on meeting Requirement 1 of GSR Part 4 [2] on a graded approach and Requirement 14 of GSR Part 4 [2] relating to the scope of the safety analysis for a PSA. Quantitative results of PSA are often used to verify compliance with probabilistic safety goals or criteria, which are usually formulated in terms of quantitative estimates of core damage frequency or fuel damage frequency, frequencies of radioactive releases of different types and societal risks, necessitating the performance of a Level 1, Level 2 or Level 3 PSA, respectively. Probabilistic safety goals or criteria do not usually specify which hazards and plant operational states are to be addressed. Therefore, in order to use the PSA results for the verification of compliance with existing probabilistic safety goals or criteria, a full scope PSA involving a comprehensive list of initiating events and hazards and all plant operational states should be performed unless the probabilistic safety goals or criteria are formulated to specify a PSA of limited scope, or alternative approaches are used to demonstrate that the risk from those initiating events and hazards and operating states that are not in the model does not threaten compliance with the probabilistic safety goals or criteria.

2.26. A major advantage of PSA is that it provides an explicit framework for the analysis of uncertainties in risk estimates. The identification of sources of uncertainty and an understanding of their implications on the PSA model and its results should be considered an inherent part of any PSA, so that, when the results of the PSA are to be used to support a decision, the impact of the uncertainties can be taken into account.

2.27. Requirement 23 of GSR Part 4 (Rev.1) [2] states:

**“The results of the safety assessment shall be used to specify the programme for maintenance, surveillance and inspection; to specify the procedures to be put in place for all operational activities significant to safety, and for responding to anticipated operational occurrences and accidents; to specify the necessary competences for the staff**

**involved in the facility or activity; and to make decisions in an integrated, risk informed approach.”**

2.28. The Level 2 PSA should be used during the lifetime of the plant to provide an input into decision making in combination with the results and insights of deterministic safety analyses, assessment of engineering safety features and considerations of defence in depth.

2.29. PSA can provide useful insights and inputs for various interested parties, such as operating organizations (management and engineering, operations and maintenance personnel), regulatory bodies, technical support organizations, designers and vendors, for making decisions on:

(a) Design modifications and plant modifications;
(b) Optimization of plant operation and maintenance;
(c) Safety analysis and research programmes;
(d) Regulatory issues.

2.30. Where the results of the PSA are to be used in support of the decision making process, a formal framework for doing so should be established (see Ref. [13]). The details of the decision making process will depend on the purpose of the particular PSA application, the nature of the decision to be made and the PSA results to be used. If numerical results from the PSA are to be used, reference values against which these results can be compared should be established.

2.31. The PSA should address the actual design or, in the case of a plant under construction or modification, the intended design or operation of the plant as part of the periodic safety reviews, which should be clearly identified as the basis for the analysis. The status of the plant can be fixed as it was on a specific date or as it will be when the agreed modifications are completed. This needs to be done to provide a clear reference point for completion of the PSA. Later changes can be addressed in the framework of the periodic safety reviews, as part of a living PSA programme, as described in paras. 2.202.20–2.23.

2.32. For a plant in the design stage, the results of PSA should be used as part of the design process to assess the level of safety. In this case, the insights gained from PSA should be considered in combination with the insights gained from the assessment of engineering safety features and deterministic safety analysis to make decisions about the safety of the plant. Decisions on the safety of the plant should be the result of an iterative process aimed at ensuring that national requirements and probabilistic safety goals or criteria are met, that the design is balanced, and that the risk is as low as reasonably achievable.

2.33. For a plant in the design stage or at a periodic safety review stage, the results of the PSA (including uncertainties, importance analysis and sensitivity studies) should be compared with the probabilistic safety goals or criteria if these have been specified in national regulations or guidelines. This should be done for all probabilistic safety goals or criteria defined for the plant, including those that address system reliability, core damage frequency, fuel damage frequency, frequencies of releases of radioactive material, health effects for the public and off-site consequences such as land contamination and restrictions on foodstuffs. If probabilistic safety criteria or goals are not defined, risk reduction possibilities can nevertheless be examined based on the Level 2 PSA results.

2.34. The PSA should aim to identify all accident sequences that contribute in a non-negligible way to risk. If the analysis does not address all significant contributions to risk (e.g. if it omits external hazards or shutdown states), then the conclusions drawn from the PSA about the level of risk from the plant, the balance of the safety features provided and the need for changes to be made to the design or operation to reduce risk might be biased. Such limitations should be acknowledged when using PSA to support decision making. The use of the full scope PSA model is therefore recommended.

2.35. The results of the PSA should be used to identify weaknesses in the design or operation of the plant as well on actions considered in severe accident management guidelines strategies. These can be identified by considering the contributions to the risk from groups of initiating events, and the importance measures[7] for SSCs and human errors. Where the results of the PSA indicate that changes could be made to the design or operation of the plant to reduce risk, the changes should be incorporated where reasonably achievable (e.g. taking the relative costs and benefits of any modifications into account).

## 3. PSA PROJECT MANAGEMENT AND ORGANIZATION

3.1. Requirement 22 of GSR Part 4 (Rev.1) [2] states that "**The processes by which the safety assessment is produced shall be planned, organized, applied, audited and reviewed**.". The recommendations on project management and organization of PSA provided in SSG-3 (Rev. 1) [4] are also applicable to Level 2 PSA and are therefore not repeated here. Only those aspects that are particularly important for Level 2 PSA are presented in this section.

### DEFINITION OF THE OBJECTIVES OF THE LEVEL 2 PSA PROJECT

3.2. Paragraphs 3.2–3.4 provide recommendations on meeting Requirement 4 of GSR Part 4 (Rev.1) [2] with regard to defining the objectives of the Level 2 PSA project. Differing end uses place differing emphases and requirements on the various inputs into, and components of, a Level 2 PSA. The objectives of the Level 2 PSA project should be set out fully at the beginning of the project and they should be in agreement with the main objective of the Level 2 PSA and purposes intended as described in Section 1.

3.3. The limitations of both Level 1 PSA and Level 2 PSA should be identified, taken into account and documented in the Level 2 PSA project, with account taken of the objectives, intended uses and applications of the Level 2 PSA.

3.4. The objectives of the Level 2 PSA project should be understandable and achievable by the users of the Level 2 PSA. In this context, gathering previous experiences on other Level 2 PSA project management is highly recommended.

[7] Typical importance measures used in probabilistic safety assessment are Fussell-Vesely importance, Birnbaum importance, risk reduction worth and risk achievement worth (described in para 5.170 of SSG-3 (Rev. 1) [4]) giving a perspective on how an individual basic event, groups of basic events, credited systems and groups of initiating events contribute to the overall risk profile.

## SCOPE OF THE LEVEL 2 PSA PROJECT

3.5. Paragraphs 3.6-3.7 provide recommendations on meeting Requirements 1 and 14 GSR Part 4 (Rev.1) [2] in relation to the scope of the Level 2 PSA project.

3.6. The scope of the Level 2 PSA project should be determined by the overall scope of the Level 2 PSA, as described in paras 2.5–2.15. The scope of the Level 2 PSA project should follow a graded approach to define the scope and the methods used for modelling the severe accident phenomena and for the contribution of the SSCs to the risk of radioactive releases depending on their source (see para 1.19). A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs being part of the installation containing potential sources of radioactive releases other nuclear power plants (e.g. fault tree and event tree development, assumptions related to human reliability analysis or equipment reliability data, fragility curves (if applicable) and reliability of digital instrumentation and control systems, including computer based systems used to control the process in the installation).

3.7. In particular, in compliance with para 2.14 when determining the scope of the Level 2 PSA project, consideration should be given to the input requirements for a Level 3 PSA, as applicable. The ultimate product of a Level 2 PSA will be a description of a number of challenges to the containment, a description of the possible responses of that containment and an assessment of the consequent releases considering the source term calculations described by the release categories definitions, frequency and characterization of their magnitude. The description will include the inventory of material released, its physical and chemical characteristics, and information on the time, energy, duration and location of the releases. Subsidiary products of the Level 2 PSA will be a description of a number of challenges to the dedicated SSCs ensuring the confinement function (e.g. the containment structure), and a description of the possible responses of those SSCs.

## PROJECT MANAGEMENT FOR LEVEL 2 PSA

3.8. Requirement 5 of GSR Part 4 (Rev.1) [2] states that "**The first stage of carrying out the safety assessment shall be to ensure that the necessary resources, information, data, analytical tools as well as safety criteria are identified and are available**." Recommendations on the decisions that PSA project managers should take and on the supervision, coordination and implementation of various tasks are provided in paras 3.3–3.14 of SSG-3 (Rev. 1) [4]. Those recommendations are also applicable to Level 2 PSA and are not repeated here. One aim of project management for Level 2 PSA is to ensure that the PSA being produced does indeed represent the plant in its 'as is' condition and reflect realistic operating practices to the extent possible, and that it does take account of recent developments in methods, models and data.

3.9. If the starting point is an existing Level 1 PSA, then coordination with the Level 1 PSA management team should be established. If the starting point is to develop jointly a Level 1 PSA and Level 2 PSA, a single management team could be established.

3.10. Although the basic framework and methods of Level 2 PSA are well established, the analysis in Level 2 PSA demands high levels of expertise and technical resources. Even when high levels of resources are employed, analyses of both the behaviour during the severe accident of the containment as well as the radiological source terms are subject to large uncertainties associated with phenomena. These aspects should be considered in accordance with the scope of the Level 2 PSA project as well as for the Level 2 PSA project management for the selection of computer codes (see paras. 3.16-3.17), selection and qualification of personnel and their trainings (paras. 3.18-3.21).

3.11. In accordance with the requirements established in [5], a management system for the project should be implemented with due consideration given to the safety implications of the results of the Level 2 PSA and its intended uses. In particular, the application of expert judgment should be justified and managed through a controlled and documented process. Provisions should be made by the Level 2 PSA project management for establishing independent review processes or performing comparative studies, as appropriate (see 3.23-3.28). Further details on the specific needs for technical review of relevant aspects of the analysis, project documentation and configuration control are provided in Section 12.

3.12. The Level 2 PSA project management should aim to ensure that the insights gained from carrying out the analysis relating to plant vulnerabilities and severe accident management are properly understood by the plant management and operating staff, so that the operating organization gains ownership of the Level 2 PSA, and by the regulatory body and other relevant interested parties.

3.13. Recommendations related to the establishment of a quality assurance programme for the development of Level 2 PSA studies as part of the duties of the Level 2 PSA project management are defined in paras 3.15–3.16 of SSG-3 (Rev. 1) [4] and are not repeated here. This quality assurance process should include activities related to the independent review performed for the Level 2 PSA (see 3.23-3.28).

3.14. Depending on the phase of the plant lifetime (e.g. design phase or operation) and on the objectives to be reached, the Level 2 PSA project management should specify and define key prior information for the successful development of the Level 2 PSA such as:

(a) Selection of staff and responsibilities (see paras 3.18-3.21);
(b) Scope and level of detail to be achieved, including the predefinition of a sufficient number of PDSs and/or of release categories (see paras 3.5-3.7);
(c) Planning and schedule of activities in the project, including the identification of needs to perform research related activities and software development, verification and validation and training;
(d) Availability and collection of plant data in relation to SSC, severe accident phenomena, human factors, emergency operating procedures and/or severe accident management guidelines and internal and external hazards (see Ref. [10]);
(e) Modelling assumptions related to the PSA (e.g. integral, mechanistic or dedicated computer codes);
(f) Procedures for using expert judgement;
(g) Definition of the format and amount of information to be presented as the Level 2 PSA results including the uncertainty and importance analysis and sensitivity studies.

## SELECTION OF SOFTWARE, APPROACHES AND METHODS

3.15. Requirement 18 of GSR Part 4 (Rev.1) [2] states that "**Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation.**" The selection of computer codes to be used for the Level 2 PSA should follow the recommendations provided in paragraphs 2.5 and 2.6 of SSG-3 (Rev. 1) [4]. In addition, specific codes for the probabilistic modelling on Level 2 PSA should be considered as advantageous to have the possibility to deal with multiple point branches as well as with correct quantification of success branches.

3.16. The selected computer codes used in Level 2 PSA should go through a process of verification and validation covering all severe accident phenomena encountered during the accident progression. Models and correlations introduced in the computer codes used for severe accident analysis for Level 2 PSA should be verified and validated by experiments and/or benchmarking to ensure a sufficient level of confidence on the results obtained and to minimize the uncertainties introduced by the simplifications and assumptions related to the physical phenomenon considered. However, it should be recognized that the level to which verification and validation can be performed for severe accident analysis codes is much lower than for other codes used to support the PSA, such as the thermohydraulic codes used to support the success criteria for safety systems in the Level 1 PSA. This is because there is, in general, a limited applicability of experimental results to real reactor conditions, as it is not always possible to carry out experiments that reflect the extreme conditions that occur in a severe accident and the scale of the geometry of the reactor coolant system and the reactor containment.

3.17. The selection of an integrated or separated approach (see para. 2.6) and the methods for accident progression event tree construction (see paras 9.1-9.17), should be consistent with the scope and objectives of the Level 2 PSA project.

TEAM SELECTION FOR THE LEVEL 2 PSA PROJECT

3.18. In the selection of the Level 2 PSA team, it should be ensured that there is an adequate level of expertise in the following areas: (i) knowledge of the design and operation of the plant, (ii) knowledge of severe accident phenomena and on challenges to the containment, and (iii) knowledge of PSA in general, and of Level 2 PSA techniques in particular. The depth of the team's expertise can be different depending on the stage in the lifetime of the plant at which the Level 2 PSA is carried out, the scope of the Level 2 PSA and the intended applications of the Level 2 PSA. To the extent possible, extensive participation of the plant engineers and utility personnel, or designers (e.g. if performed at the design stage), and probabilistic safety analysts specialized in accident phenomena and other Level 2 PSA disciplines is essential.

3.19. The Level 2 PSA project management should provide working arrangements that ensure that there are good interactions and communication between all the members of the Level 2 PSA team, including project managers and analysts. In addition, the Level 2 PSA project management should aim to ensure that, as the analysis progresses and insights are developed, the approaches to the different technical areas are modified as necessary to ensure that the analysis is progressing in a coherent way and that there is a reasonable balance of efforts across all topics. The need to sustain good communication among the Level 2 PSA team during the entire Level 2 PSA project cannot be overemphasized.

3.20. Regarding the knowledge of PSA in general and of Level 2 PSA techniques in particular, the Level 2 PSA team:

(a) Should have adequate training on the computer code(s) used;
(b) Should have a sound knowledge of the models and methods implemented in the computer code(s);
(c) Should have sufficient understanding of the limitations of the computer code(s) in relation to the phenomena to be modelled;
(d) Should be familiar with the guidance on the computer code(s) and the procedures for

implementing the models into them;

(e) Should have adequate capacity to evaluate the results of the computer code(s).

3.21. For a nuclear power plant in operation, the Level 2 PSA team should consider including:

(a) A technical project leader responsible for coordination among all the project experts;
(b) Experts in the design and operation of the plant (particularly of the containment systems), the emergency operating procedures and the severe accident management guidelines.
(c) Experts in severe accident phenomena, performance of the containment, uncertainties associated with severe accidents, chemical and physical processes governing accident progression, loads generated over the containment, releases of radionuclides and computer codes for the analysis of severe accidents;
(d) Experts in the structural design, the pressure capacity and the failure modes of the containment;
(e) Experts in developing event tree analysis, fault tree analysis, human reliability analysis, uncertainty analysis, and statistical methods all in particular for Level 2 PSA;
(f) Experts in processes for expert elicitation and judgement for Level 2 PSA computer codes and Level 1 PSA.

INDEPENDENT VERIFICATION

3.22. Requirement 21 of GSR Part 4 (Rev.1) states that "**The operating organization shall carry out an independent verification of the safety assessment before it is used by the operating organization or submitted to the regulatory body.**"

3.23. The main objective of the independent verification of the Level 2 PSA is to confirm that selected methods and approaches, probabilistic and deterministic models used, and assumptions and data considered have been applied in an adequate manner to meet the applicable safety objectives and requirements regarding the PSA. It is considered good practice to conduct an internal independent verification, as a quality assurance process integrated in the project management, during the process of development the Level 2 PSA studies. This internal independent verification, if conducted, should look at each step of development (data, computers codes used, interface with Level 1 PSA, supporting studies, assumptions in the event trees, event tree quantifications, and results interpretation) to ensure a sufficient quality of a Level 2 PSA for its uses. This internal independent verification process may help identifying some sources of uncertainties (e.g. see 5.13, 6.24-6.27, 7.23-7.30, 8.18-8.22. and 11.21-(4)).

3.24. Since the development of the Level 2 PSA and the design may be conducted in parallel as part of the iterative design process of the nuclear power plant, the licensee organization should carry out an independent verification (e.g. peer review) to ensure that Level 2 PSA results only relate to the design and operation of the nuclear power plant as submitted to the regulatory authority for approval (i.e. according to the scope of the document to be submitted to the regulatory authority), and comply with relevant regulatory requirements related to reference values and risk metrics for Level 2 PSA.

3.25. The independent verification of the Level 2 PSA performed by or on behalf of, the licensee organization should be conducted by a different group of experts or institution from those who develop the Level 2 PSA (e.g. external group or institution of the licensee organization, sometimes from a different State, to ensure that the Level 2 PSA conforms to current, internationally recognized good practices in Level 2 PSA). The licensee organization should ensure the independence between the organization conducting the Level 2 PSA and the group of experts or the institution in charge of the independent verification.

3.26. The Level 2 PSA comprises a number of analytical methods and approaches to model complex phenomena with their associated uncertainties, based on computational tools that might have limited resources for validation and use of expert judgement. Therefore, the licensee organization should ensure the adequate level of expertise of the group of experts or institution in charge of the independent verification to ensure the adequate evaluation of the data, assumptions and models (i.e. deterministic and probabilistic) considered in the Level 2 PSA (see para 3.28).

3.27. The results of the independent verification of the Level 2 PSA should be reported in a separate document to be presented, upon request, to the regulatory authority.

3.28. The report compiling the results of the independent verification of the Level 2 PSA should consider the assessment of the appropriateness and comprehensiveness of:

(a) The PDSs development, grouping and quantification;
(b) The analysis of accident progression and the associated systems;
(c) The models of phenomena that could occur in relation to the behaviour of the containment of the nuclear power plant following core damage;
(d) The accident progression event tree models and supporting models as well as the methods for solution of the logic models;
(e) The probability development (e.g. phenomena probabilities based on data or expert judgement);
(f) The release categories development, grouping, quantification and source term characterization;
(g) Supporting calculations, correct and appropriate application of codes;
(h) Structural analysis/fragility curves;
(i) The models for considering the human reliability analysis;
(j) The consideration of equipment reliability taking into account the equipment qualification or survivability (in particular for severe accidents scenarios);
(k) The uncertainties and sensitivity analysis carried out (e.g. the bases for the selection of probability distributions of uncertain parameters and assignment of their distributions parameters).

# 4. FAMILIARIZATION WITH THE PLANT DESIGN AND SEVERE ACCIDENT MANAGEMENT

## IDENTIFICATION OF DESIGN ASPECTS IMPORTANT TO SEVERE ACCIDENTS

4.1. This section provides recommendations on meeting Requirements 6–13 of GSR Part 4 (Rev.1) [2] for Level 2 PSA. Before starting the analysis, the Level 2 PSA team should become familiar with the design and operation of the plant. The aim should be to identify and highlight plant SSCs and operating procedures that can influence the progression of severe accidents, the response of the containment and the transport of radioactive material inside that containment. Design features that can influence the progression of a severe accident and Level 2 PSA include: fan coolers, containment sprays and/or filtered containment venting systems and suppression pools. This exercise should include the reactor building and/or the auxiliary building and the secondary containment or other relevant structures and buildings which all depend on the reactor technology and design. For existing plants, familiarization with the plant should include a plant walk-through and should involve the participation of operating personnel and plant technical staff. The plant familiarization should involve all members of the Level 2 PSA team.

4.2. The specific reactor technology and plant features that might influence the progression of a severe accident should be identified and characterized. Examples of the features that should be identified for light water reactors are as follows:

(a) The area under the reactor pressure vessel is important with regard to the behaviour of molten core material after it exits the bottom of the reactor pressure vessel, since the area influences the extent to which the molten core material will spread and its coolability;
(b) The flow paths from the area under the reactor pressure vessel to the main containment volume. Restrictions to the flow or other geometric aspects of the flow path will reduce the extent to which core debris is dispersed following a lower head failure. This is particularly important for high pressure melt ejection in a light water reactor;
(c) A highly compartmentalized containment configuration will limit the extent to which combustible gases mix and become distributed in the containment atmosphere;
(d) Features that could lead to containment bypass sequences.

These and other plant specific design features should be identified for further investigation.

4.3. Examples of key design features of the plant that are significant in respect of the progression and mitigation of severe accidents are listed in TABLE 1. In addition to plant features, relevant operating procedures and severe accident management guidelines should also be considered.

TABLE 1. EXAMPLES OF KEY PLANT AND/OR CONTAINMENT DESIGN FEATURES

| Key plant and/or containment design feature | Comment |
|---|---|
| REACTOR | |
| Reactor type | Boiling water reactor, pressurized water reactor, advanced gas cooled reactor or other |
| Power level | Total thermal power at steady state |
| Type of fuel mix/type of cladding | Oxide, mixed oxide/zircaloy, stainless steel, ceramic or other |
| CORE | |
| Mass of fuel and mass of cladding | Actual operational values |
| Fuel assembly geometry | Actual operational values |
| Type and mass of control rods | Actual operational values |

| Key plant and/or containment design feature | Comment |
|---|---|
| Spatial distribution of reactor power | Typically axial and radial peaking factors |
| Decay heat | Total decay heat level as a function of time |
| Radioactive material inventory | Full inventory of radionuclides in the core |
| REACTOR COOLANT SYSTEM | |
| Reactor coolant and moderator types | Water, heavy water, $CO_2$, helium and others |
| Reactor coolant system coolant/moderator volume | As designed and fabricated |
| Accumulator volume and pressure set point and number | Actual operational values for each type of accumulators |
| Reactor coolant system depressurization devices and procedures | Specify set point and procedures |
| Pressure relief capacity | Actual operational values |
| Isolation of containment penetrations connected to the reactor coolant system | Potential for containment bypass |
| Safety systems actuation mechanism | Passive or active |
| Safety systems injection volume and pressure set point | Actual operational values |
| CONTAINMENT [a)] | |
| Containment geometry | Shape and separation of internal volumes |
| Containment free volume | As built, taking into account displacement by structures |
| Containment design pressure and temperature | A realistic assessment of maximum capacity is needed for the PSA |
| Containment design leakage and conditions of leakage | Actual operational values |
| Containment material construction | Steel, concrete, other |
| Operating pressure and temperature | Actual operational values |
| Hydrogen control mechanisms | Provision of inertness, ignitors, passive recombiners, other |

| Key plant and/or containment design feature | Comment |
|---|---|
| Suppression pool volume | Water and atmosphere volumes |
| Containment cooler capacity and set points | Actual operational values |
| Concrete aggregate of each containment structures | Specify chemical content |
| Design of cavity, keyway or pedestal | Dispersive versus non-dispersive |
| Flooding potential of cavity or pedestal | Flooded or dry |
| Sump(s), volume filters and location(s) | Geometric details, identification of materials (painting, pipe insulation, etc.) potentially affecting sump filter clogging |
| Heat removal paths from reactor and containment | Layout, location and operation |
| Configuration of heat sink | Operational procedure |
| In-containment refuelling water storage tank or refuelling water storage tank or other in-containment water storage tank | Location, volume and operation |
| Proximity of containment boundaries | Distance from reactor pressure vessel and cavity or pedestal |
| Containment venting procedure and location | Location of vent line and actuation procedure |
| Response to external hazards | Structural damage due to seismic events or flooding events or transportation events |
| Potential for containment isolation failure | Penetration arrangements and reliability of seal materials for containment isolation |
| Potential for cooling of molten core | Design of Generation III+ plants includes some features for cooling of the spread molten core |
| SPENT FUEL POOL | |
| Geometry | Shape, separation into sections, specific layout |
| Capacity and arrangement | Number of maximum and actual stored spent fuel assemblies, racks design, loading pattern (if any) |
| Decay heat | Total decay heat at normal storage conditions and for emergency unloaded core |

| Key plant and/or containment design feature | Comment |
|---|---|
| Radioactive material inventory | Full inventory of radionuclides in the fuel stored in the spent fuel pool |
| Design parameters | Nominal coolant temperature and level, maximum allowed coolant temperature, minimum allowed coolant level |
| Safety features | Nominal and minimum flow rate, coolant inventory, and soluble absorber concentration, nominal and maximum temperature of flow rate |
| Materials and composition | Steel, concrete, other |

[a] The specific information listed here might change in some areas for plants without a pressure retaining containment (e.g. nominal leak rate will need to be included for plants with structures that provide a confinement function) or with a different type of containment.

CONSIDERATIONS REGARDING MULTIPLE UNITS OR MULTIPLE RADIOACTIVE INSTALLATIONS ON A SITE

4.4. Paragraphs 4.5 to 4.9 aim at providing an overview of key aspects to be identified from the plant familiarization perspective when performing a Level 2 PSA for a site where multiple units or installations with radioactive sources are located. Detailed recommendations on Level 2 PSA for multiple unit sites are provided in Section 14.

4.5. Site organizational aspects should be identified and recognized as an important aspect affecting the modelling of Level 2 PSA for multiple unit sites.

4.6. Performance shaping factors considered for the human reliability analysis should consider conditions related to field operations and environmental conditions when several units are at different stages of the severe accident progression or induced by the impact of internal and external hazards and their combinations (see Ref. [10]).

4.7. Considerations related to shared equipment among units on the site, either installed or non-permanent, should be identified.

4.8. Potential common cause failures among similar equipment at different units should be identified for the purpose of the development of the Level 2 PSA model for multiple unit sites.

4.9. The availability, capability and accessibility of the ultimate heat sink and of electrical power supply sources for multiple units on a site should be considered.

REVIEW OF STRATEGIES TO COPE WITH SEVERE ACCIDENT ASSOCIATED PHENOMENA

4.10. Paragraphs 4.11 to 4.15 aim at providing an overview of key aspects to be considered from the plant familiarization perspective in relation to strategies to cope with severe accident associated phenomena when performing a Level 2 PSA.

4.11. For the plant familiarisation, the analyst should collect available documentation on the strategies implemented at the plant and become familiar with the priorities and actions contained within these strategies. Strategies developed to cope with severe accident progression generally include those aimed at (1) protecting the confinement function, including preventing the containment bypass, (2) if applicable, protecting the reactor building where the spent fuel pool is located. Depending on the reactor design, strategies should also address protection of the proper functioning of filtered venting systems in auxiliary building and management of leakage of liquid effluent from reactor containment in case of recirculation of contaminated water outside the containment. During the progression of a severe accident of the fuel in the reactor vessel (e.g. in the reactor core for water cooled reactors), two important strategies are considered, firstly, in-vessel cooling and retention of damaged fuel (e.g. in-vessel melt retention for some reactor technologies such as water cooled, metal cooled and molten salt) and, secondly, ex-vessel cooling and retention of damaged fuel (e.g. ex-vessel corium[8] cooling for some water cooled reactor designs). See also paras 4.14–4.15.

4.12. The analyst should also be familiar with other strategies related to severe accident management. These strategies may include the spent fuel pool and the long term phase of the severe accident, such as the control of combustible gases in the atmosphere of the containment, the control of the pressure inside the containment and the control of radioactive releases from the containment. IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [14] provides recommendations on the long term phase of the severe accident related to the use of non-permanent equipment; the maintenance and inspection of non-permanent equipment; waste management due to long term actions such as water treatment; limits on dose rates to ensure the operator actions; and availability of electricity, compressed air or water sources.

4.13. Paragraphs 0-4.15 provide recommendations on relevant information on safety provisions[9] that should be collected in the familiarization task for the Level 2 PSA related to the success of strategies to deal with core damage[10].

**In-vessel melt retention**

4.14. For water cooled reactors, the in-vessel melt retention strategy is aimed at ensuring a passive and/or active reflooding of the reactor pressure vessel cavity up to a level to ensure and maintain, with sufficient confidence, the integrity of the reactor pressure vessel by cooling it from outside and the integrity of the corium inside by in-vessel water. For other reactor

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[8] Corium is a complicated mixture of fuel, zirconium alloy and steel, which forms in water cooled reactors as a result of thermochemical reactions, including between zirconium and water.

[10] The term ‘core damage’ is equivalent to ‘core melt’ for some reactor technologies (e.g. water cooled reactors).

technologies, in-vessel melt retention might be defined slightly differently, depending on the specifics of that reactor technology (e.g. non-pressurized reactor coolant system). Relevant information to be collected in the familiarization task related to the success of this strategy could be on, for example:

(a) Design of safety provisions for passive and/or active reflooding of the reactor cavity;
(b) Design of safety provisions for in-vessel water injection, where applicable (e.g. to reduce the risk of the focusing effect[11]);
(c) Reactor pressure vessel insulation (e.g. for water cooled reactors, to consider first sufficient water circulation between the reactor pressure vessel wall and the insulation and second a path for the evacuation of the produced steam to upper volume of the containment);
(d) Design solutions related to reactor pressure vessels internals (e.g. large mass of steel in lower plenum may reduce the risk of the focusing effect at the reactor pressure vessel wall);
(e) Reactor pressure vessel design (e.g. geometry, thickness and low bottom reactor pressure vessel penetrations);
(f) Water inventory available (i.e. affecting the delay the time of corium arrival in the lower plenum and therefore reduce the residual power to extract).

**Ex-vessel corium cooling**

4.15. For water cooled reactors, ex-vessel corium cooling can consist of a passively controlled and gradual spreading of the corium outside of the reactor pressure vessel on a surface, allowing effective corium cooling by passively and actively injecting water up and down the corium layer. Relevant information to be collected in the familiarization task related to the success of this strategy could be, for example:

(a) Analysed reactor vessel failure modes to support the strategy;
(b) Relevant operator actions to support the strategy;
(c) The configuration of the corium spreading surface (relevant to reduce downward heat flux to the concrete and hence reducing its ablation);
(d) Potential sequences for a wet reactor cavity (relevant to the risk of steam explosion);
(e) Functional criteria (e.g. timing, volume, automatic or manual actuation) for safety provisions needed for residual heat removal from the corium spreading surface (e.g. passive or active or a combination of both);
(f) The chemical composition of materials in the corium spreading surface (e.g. provisions to reduce the risk of recriticality or to facilitate the corium cooling).

COLLECTION OF INFORMATION IMPORTANT TO SEVERE ACCIDENT ANALYSIS

4.16. Requirement 19 of GSR Part 4 (Rev.1) [2] states that “**Data on operational safety performance shall be collected and assessed.**” When the PSA team has developed a general

[11] The ‘focusing effect’ phenomenon depends on the metal composition of the layers formed when the mix of melted fuel and the internal structures are relocated in the bottom of the reactor pressure vessel. The focusing effect could be understood as the behaviour of the metal layer and impact of its thickness on heat flux affecting the integrity of the reactor pressure vessel wall.

understanding of the plant design, phenomena[12] and features that may influence severe accidents and releases of radioactive material, the quantitative data that are necessary to carry out the plant specific analysis should be collected and organized. The data necessary for the PSA depend in part on the scope of the analyses and the nature of the computational tools. For example, the amount and type of input data collected may depend on the plant specific computer model used to calculate accident progression. Detailed architectural and construction data for the containment structure should be collected to develop plant specific model calculations of the containment performance if such calculations are required by the scope of the containment performance analysis.

4.17. Data should be obtained from sources, such as:

(a) Design documents and/or plant licensing documents, such as safety analysis report, technical specifications, system(s) descriptions;
(b) As built drawings;
(c) Plant specific normal operating, maintenance or test procedures;
(d) Information on plant automatic actuations;
(e) Emergency operating procedures and severe accident management guidelines;
(f) Engineering calculations or analysis reports;
(g) Observations during plant walkdown reports and/or walkdown reports;
(h) Construction standards;
(i) Regulatory requirements;
(j) Vendor manuals;
(k) Other relevant plant documents.

References to the source(s) of data should be recorded as part of the PSA documentation.

4.18. If the intent is to use data from a reference plant in the development of the Level 2 PSA, the plant specific data should be compared with reference plant values. Such a comparison is of great value in determining whether the two plants are in fact similar and therefore would likely have similar vulnerabilities. TABLE 2 lists examples of design features of the plant and containment of water cooled reactors for comparison with those of other plants and how they can be used. However, great care has to be applied when drawing conclusions from such a comparison.

TABLE 2. SAMPLE COMPARISON OF PLANT AND CONTAINMENT DESIGN CHARACTERISTICS OF WATER COOLED REACTORS

| Parameter and design feature | Significance or comparability |
|---|---|
| Ratio of reactor power to reactor coolant system volume | Accident progression times, time for recovery actions |
| Ratio of reactor power to containment volume | Scaling of containment loads |

[12] Source of information for the phenomena could be obtained from the Phenomena Identification and Ranking Table (PIRT) analysis for severe accidents, if available.

| Parameter and design feature | Significance or comparability |
|---|---|
| Ratio of zirconium mass to containment free volume | Potential for combustion and scaling of containment loads |
| Core catcher | Spreading area, corium cooling devices |
| Vessel cavity to containment pathways | Potential for melt ejection and dispersal in containment at high pressure |
| Concrete aggregate (composition) | Non-condensable gas generation and radioactive material release during molten core–concrete interaction; efficiency of corium cooling by water submersion. |

## 5. INTERFACE WITH LEVEL 1 PSA: GROUPING OF SEQUENCES

5.1. This section provides recommendations on the interface between Level 1 PSA and Level 2 PSA. It addresses the analysis of results and information from the Level 1 PSA that need to be carried out to provide the necessary input for the Level 2 PSA. The detailed implementation of Level 1 PSA and Level 2 PSA interface will depend on methodology chosen for the Level 2 PSA, the modelling software used and the reactor technology.

5.2. According to SSG-3 (Rev. 1) [4], Level 1 PSA identifies a large number of accident sequences that lead to core damage. It is neither practical nor necessary to individually treat each accident sequence when assessing accident progression, containment response and radionuclide release within Level 2 PSA. Accident sequences should be grouped together into PDSs in such a manner that all accidents within a given PDS can be treated in the same way for the purposes of the Level 2 PSA. If necessary, the accident sequence models in the Level 1 PSA should be adjusted to take account of the specific needs of the Level 2 PSA. PDSs should represent groups of accident sequences that have similar accident timelines, containment status and containment system (un)availability status and which generate similar loads on the containment, thereby resulting in a similar event progression and similar radiological source terms. Attributes of accident progression that will influence the chronology of the accident, the progression of the core damage, the containment response or the release of radioactive material to the environment should be identified. The attributes of the PDSs provide boundary conditions for the performance of severe accident analysis.

5.3. Proper care should be taken to ensure that optimisms are not introduced when sequences from the Level 1 PSA are mapped and transferred to the Level 2 PSA and that no sequences are lost or duplicated. The latter can be achieved by quantifying the frequencies of all the PDSs and validating their sum against the core damage frequency determined from Level 1 PSA (typically core and/or fuel damage frequency). Justifications for any numerical deviations should be given.

5.4. The success criteria for system modelling in PSA should specify the system mission time to reach the safe state or to fulfil the modelled system function. In particular for Level 2 PSA, the mission time should be defined adequately for capturing the severe accident progression, the time needed for design features to effectively cope with severe accidents, including possible cliff-edge effects, and to ensure that the residual risk accrued after the mission time is negligible.

5.5. The following subsections provide examples of the attributes that may need to be taken into account in defining PDSs. Examples of such attributes for water cooled reactors are given in Table 3.

TABLE 3. EXAMPLES OF ATTRIBUTES OF PLANT DAMAGE STATES FOR WATER COOLED REACTORS

| | |
|---|---|
| Initiating event | Large loss of coolant accident<br>Small loss of coolant accident<br>Safety or relief valve stuck open<br>Transient, such as:<br>Reactor trip<br>Loss of off-site power<br>Loss of electrical bus<br>Loss of feedwater<br>Loss of service water<br>Steam line break<br>Feedwater line break<br>Anticipated transient without scram<br>Bypass event (loss of coolant accident in interfacing system or steam generator tube rupture)<br>Reactivity accidents (e.g. homogenous or heterogenous dilution or control rods accidental withdrawal / ejection) |
| Reactor coolant system status | Reactor coolant system pressure at core damage:<br>— High (relief valves are challenged)<br>— Medium (above low pressure coolant injection head)<br>— Low (including method of depressurization)<br>— Status of safety relief valves<br>Reactor coolant system integrity (shutdown states):<br>— Vessel head removed<br>— Nozzle dams installed<br>— Safety valves removed<br>— Vent open<br>Reactor coolant system inventory (shutdown states):<br>— Full power inventory<br>— Flooded refuel cavity<br>— Mid-loop operations in a pressurized water reactor<br>State of fuel in the reactor for decay heat (shutdown states):<br>— Pre/post refuelling<br>— Time since reactor shutdown |

| | |
|---|---|
| Status of emergency cooling system and other cooling systems (timing of core damage and ability to prevent further core damage progression) | All injection fails to start (no injection, early damage)<br>Coolant injection initially successful, but recirculation cooling fails (later core damage)<br>Emergency core cooling functionality after core damage or breach of reactor pressure vessel<br>Steam generator cooling availability |
| Status of containment's engineered safety features | Sprays (if any):<br>— Operate at all times<br>— Fail on demand<br>— Initially operate, but fail on switchover to recirculation cooling<br>Suppression pool (if any):<br>— Effective at all times<br>— Ineffective (pool drained or bypassed early)<br>— Bypassed late<br>Fan coolers (if any):<br>— Operate at all times<br>— Fail on demand<br>— Fail late<br>Venting systems:<br>— Operate at all times<br>— Fail on demand<br>— Fail late<br>Status of containment inerting systems (if any)<br>Status of hydrogen control systems<br>Containment passive heat removal system (if any):<br>— Available<br>— Unavailable<br>— In operation<br>— Failed |
| Status of support systems | Electrical alternating current and direct current power<br>Component cooling<br>Instrument Air<br>Heating/ventilation and air conditioning<br>Availability/accessibility of mitigating systems |
| Containment status | Intact and isolated at the onset of core damage<br>Intact, but not isolated at the onset of core damage<br>Structural failure or enhanced leakage (with indication of size and location of leakage)[a] |
| Status of secondary containment (reactor | Intact and isolated at the onset of core damage |

| | |
|---|---|
| building or enclosure building) | Intact, but not isolated at the onset of core damage<br>Structural failure or enhanced leakage[a] |

[a] This includes any external events that may damage containment structures

PLANT DAMAGE STATES FOR PSA FOR INTERNAL INITIATING EVENTS DURING FULL POWER CONDITIONS

5.6. If the Level 2 PSA is developed following a separated approach (see para 2.6) Level 1 PSA does not account for specific aspects relevant to the specification of PDSs. For example, the Level 1 PSA may not have addressed the status of containment systems or other systems that do not directly affect the determination of core damage (i.e. they do not contribute to the success criteria for preventing core damage). In such cases, the Level 1 PSA should be expanded to take into account the missing aspects in the specification of PDSs (see Table 3 for reference). One method for incorporating such missing systems into the Level 1 PSA is to develop bridge trees that link to Level 2 PSA system models, as shown in FIG. 1, thereby capturing important dependencies (support systems, operator performance, etc.).

5.7. If the Level 2 PSA is developed as part of an integrated Level 1 – Level 2 PSA (see para 2.6), many of the PDS characteristics listed later in paras. 5.10-5.12 will be implicitly available for the Level 2 PSA model. Such an approach may allow to reduce the number of PDS needed. In any case, even though the structure of the PDSs could be simpler in an integrated Level 1 PSA and Level 2 PSA model, the analyst should verify that simplifications or assumptions in Level 1 PSA model will not screen out possible PDSs contributing to radioactive releases.

5.8. Generally, PDSs can be divided into two main classes: those in which radioactive material is released from the reactor coolant system to the containment and those in which the containment is either bypassed or is ineffective at the time of core damage. Thus, the PDSs should specify the containment status (e.g. intact and isolated, intact and not isolated, failed or bypassed) and, for PDSs where the containment is bypassed, should specify the type and size of the bypass (e.g. loss of coolant accident in interfacing systems, steam generator tube rupture). If the reactor building or secondary containment is likely to have a major influence on the source term, then its status at the time of core damage is specified by means of the PDS. For PDSs with intact containment, an accident progression event tree analysis should be performed. For PDSs where containment is failed or bypassed, only a source term analysis may be necessary, though a simplified event tree may need to be provided in the model. However, for some accidents an accident progression event tree may be needed to address possible plant features that can reduce the source term (e.g. scrubbing of fission products by means of a water pool or water spray, actions for reducing or isolating the containment bypass) or to quantify possible containment additional damages that increase the releases.

5.9. The characteristics specified for the PDSs are generally left to the discretion of the analyst. Examples of characteristics are given in paras. 5.10-5.12. It should be noted that the level of detail of characteristics used to define the PDSs depends on the case used for the development of Level 1 PSA and Level 2 PSA (see para 2.6). If the Level 2 PSA is developed as an extension of Level 1 PSA, the definition and selection of characteristics specified for the PDSs should be justified.

**Plant damage states without containment bypass**

5.10. In specifying PDSs without containment bypass, account should be taken of the equipment and system failures identified within Level 1 PSA that could affect either the challenge to the containment or the release of radioactive material. Depending on the reactor technology, examples are the following (see Table 3 for further details for water cooled reactors):

(a) Type of initiating event, which can, for example, affect the rate of discharge of reactor coolant in the containment, the progression of the core damage and of hydrogen generation, and the timing of the release of radioactive material.
(b) Failures of the credited systems (e.g. reactor protection system, residual heat removal system or emergency core cooling system) that have occurred, leading to core damage;
(c) Extent of fuel damage.
(d) The time at which core damage occurs (e.g. early or late relative to the time of reactor trip).
(e) The reactor coolant system pressure at the onset of core damage and the status of safety valves or relief valves and other components that could change the pressure in the reactor pressure vessel before failure of the lower head of the reactor pressure vessel.
(f) The pressure in the reactor pressure vessel at the time of lower head failure may affect the mode of discharge of debris to the containment. This, in turn, could present a challenge to containment integrity if, for instance, high pressure melt ejection and direct containment heating ensue.
(g) The pressure in the reactor pressure vessel after the onset of core damage also affects the possibility of temperature and pressure induced failures of the reactor coolant system (e.g. creep rupture of piping and steam generator tubes, or thermal seizure of a safety or relief valve in the open position). The pressure will be affected by the initiating event and the functionality of any depressurization system.
(h) The integrity of the containment (e.g. intact, failed, isolation failure, bypassed due to a steam generator tube rupture (in a pressurized water reactor) or a loss of coolant accident at interfacing systems).
(i) Loss of coolant accident with or without pressure suppression capability (e.g. for a boiling water reactor).
(j) The state of the suppression pool (e.g. subcooled or saturated) when core damage occurs (for a boiling water reactor).
(k) The availability of the containment protection systems (e.g. containment sprays, heat removal systems and hydrogen mixing or recombiners).
(l) Initial and boundary conditions including the availability of alternating current and direct current power and associated recovery times.
(m) The actions by operating personnel that have been attempted and failed.

5.11. The status of the engineered safety features[13] of the containment is of high importance in determining the response of the containment, and such safety features should be taken into account in the grouping of accident sequences into PDSs, as they may influence processes such

[13] The attributes listed in Table 3 should be adjusted, as appropriate, for plants with structures that provide a confinement function rather than pressure retaining containments.

as cooling, the removal of radioactive material, or the mixing of combustible gases in the containment. Other attributes of PDSs may be important in some applications of PSA. For instance, if the PSA is being used to help determine accident management measures, then the status of the electrical power supply should be taken into account, since this information may be needed for some later actions. The details on how these characteristics are taken into account may depend on the methodology used for linking Level 1 and Level 2 PSA, although these issues should be addressed irrespective of the methodology applied.

**Plant damage states with containment bypass**

5.12. For PDSs with containment bypass, the main consideration should be the identification of attributes that are associated with attenuation of concentrations of radioactive material along the release pathway or affect the timing of release. This should include the type of initiating event, the status of the emergency core cooling system (including failure time), if the leak pathway is isolable after a given time period, or whether scrubbing of fission products can be justified (e.g. scrubbing in the secondary side during a steam generator tube rupture accident or scrubbing through a flooded auxiliary building during an interfacing system loss of coolant accident)[14]. For leaks into the auxiliary building or an equivalent one, the status of emergency exhaust filtration systems, heating, ventilation and air conditioning, or whether or not the leak is submerged, could be significant and should be taken into account.

**Final selection of plant damage states**

5.13. If the consideration of all factors and parameters that affect the Level 2 PSA results in a too large number of potential PDSs, then they should be reduced to a manageable number. Two approaches could be used. The first is to combine similar PDSs and perform a bounding analysis to select a representative sequence that characterizes the PDS for the purpose of the Level 2 PSA. For instance, if the Level 2 PSA relies on time consuming physical calculations, it could be possible to run a manageable number of these calculations and attribute the outcomes of one calculation to several PDSs which are similar in regard of the accident progression. This could allow to deal with a large amount of PDSs without running a non-manageable number of physical calculations. The second approach is to use a frequency cut-off as a means of screening out less important PDSs[15]. The analyst should justify that the screening carried out does not introduce a significant underestimation of the risk calculated by the Level 2 PSA; careful evaluation is necessary prior to introducing a frequency screening criterion at the PDS level. This is especially true when dealing with PDSs that could involve an early radioactive release or a large radioactive release of radionuclides to the environment. In any case, in the selection process account should be taken of the degree of variability and uncertainty introduced in the Level 2 PSA by the grouping of accident sequences into PDSs and consideration should be given to how this affects the specific objectives of the PSA.

[14] Isolation of containment bypass might not prevent severe accident progression scenarios.

[15] In some Member States a cut-off value in terms of percentage of the total risk metric (Large Release Frequency or Large Early Release Frequency) is established to consider significant PDSs from less important PDSs.

## PLANT DAMAGE STATES FOR LOW POWER AND SHUTDOWN MODES OF OPERATION

5.14. Differences in the Level 2 PSA with respect to the mode of operation and power level when the initiating event occurs, result primarily from differences in the reactor coolant inventory and in the status of both the reactor coolant system and the containment containing the reactor (e.g. reactor containment building). The PDSs specified for full power conditions should be reviewed and adapted for low power and shutdown modes; direct use of PDSs specified for Level 2 PSA for full power conditions may not be possible.

5.15. Additional PDSs, different from those for a PSA for full power operation, should be specified in a Level 1/Level 2 PSA interface for a low power and shutdown PSA to capture the unique conditions that could have a major impact on plant behaviour in severe accidents. For example, additional PDSs may be necessary for conditions unique to certain shutdown states such as those with the reactor vessel head removed or with the containment hatch opened. The following additional accident sequence characteristics should be considered in specifying the PDSs for low power and shutdown (see also SSG-3 (Rev. 1) [4]):

(a) Decay heat level (time since shutdown from power operation);
(b) State of the containment – especially when it is open;
(c) Conditions that determine the time to restore the isolation of the containment and its potentially reduced effectiveness (leaktightness) during such time;
(d) The integrity of the reactor coolant system pressure boundary with reactor vessel head removed, nozzle dams installed, safety valves removed, reactor coolant system vent opened;
(e) The coolant inventory in the reactor coolant system;
(f) Closure status of the containment and associated manual actions to close it prior to core damage.

## CONSIDERATIONS FOR INTERNAL AND EXTERNAL HAZARDS IN LEVEL 2 PSA

5.16. In order to extend the scope of Level 2 PSA to include internal and external hazards such as fire, seismic and external flooding, the potential impact of the hazards on systems necessary for mitigation of severe accidents, including systems that support operator actions, as well as the impact on the integrity of the containment, should be taken into account (see Ref. [10]), if those aspects have not yet been taken into account in the Level 1 PSA output. For example, in a Level 2 PSA for fire, the cables associated with the systems ensuring the confinement function would need to be tracked to assess impact from fire scenarios; or operator actions in the main control room may need to be failed when the main control room is assumed abandoned. In a Level 2 PSA for seismic events, the impact of the seismic event on the core cooling system, water storage tanks, systems ensuring the confinement function, other mitigating equipment, etc. should be taken into account. This could lead in some cases to the specification of a new set of distinct PDSs. The analyst should consider the need to introduce new PDSs and possibilities for assimilating new PDSs into existing ones; for instance, some failures of the systems ensuring the confinement function could be assimilated into already defined isolation failures for systems ensuring the confinement function.

5.17. In addition to para 5.16, the potential impact of hazards on the systems ensuring the confinement function[16] should be taken into account as part of the Level 2 PSA, if those aspects have not yet been taken into account in the Level 1 PSA output.

5.18. Depending on the analyst choice, human actions that occur before or soon after core damage may be credited in the Level 1 PSA and captured as part of an attribute to the PDSs for the Level 2 PSA as described in para. 8.4. When extending the Level 2 PSA for other internal and external hazards, the environmental or physical conditions introduced by the hazards may interfere with these human actions and therefore should be taken into consideration when specifying the PDSs for other internal and external hazards.

5.19. The analysis process to be conducted for considering hazards and their combinations for Level 1 PSA is described in paras 6.1-6.27 of SSG-3 (Rev. 1) [4]. This process is applicable to Level 2 PSA and it is not repeated here. For the Level 2 PSA, single as well as combined hazards have the potential to result in accident sequences induced by common cause initiators that might impact the confinement function. Some examples are:

— A design extension condition earthquake resulting in a station blackout and a containment failure, perhaps with consequential internal fire or flooding;
— A combination of external flooding and high winds hazards that might lead to the loss of the heat sink together with the loss of off-site power;
— An aircraft crash causing a common loss of offsite power and emergency diesel generator failure, which again does not only result in a plant transient but an accident sequence with containment bypass and releases of radionuclides (airborne or via water path).

5.20. In order to be widely applicable, the Level 2 PSA for hazards should be based on a full scope Level 1 PSA covering hazards as described in SSG-3 (Rev. 1) [4]. This requires that the Level 1 PSA:

(a) Does not only include a comprehensive set of internal initiating events, but also a set of relevant internal and (natural and human induced) external hazards including combined hazards as defined in SSG-64 [6] and SSG-3 (Rev. 1) [4];
(b) Covers all plant operational states. This will ensure that the insights from the PSA relating to the risk significance of accident sequences, SSCs, human errors, common cause failures, etc. are derived from a comprehensive, integrated model of the plant.

5.21. It should be noted that the development of a Level 2 PSA for hazards depends on the scope set for the Level 2 PSA but can also be influenced by the L1 hazards PSA results. In particular, in case of a low strength of knowledge associated to the Level 1 PSA results, the relevance of extending this PSA to Level 2 should be analysed with regards to safety issues, feasibility and ease of analysing insights from it.

[16] Typical examples of impacts from hazards are failures of the isolation function of systems ensuring the confinement function due to internal fire, explosion or flooding at the plant, damage of the containment due to seismic events, aircraft crashes or external explosions (blasts).

5.22. If the Level 2 PSA is based on a Level 1 PSA with a more limited scope or details, these limitations need to be taken into account in the application of the Level 2 PSA.

5.23. Those hazards, single as well as combined ones, which were screened out from further (bounding or detailed) analysis within the Level 1 PSA should also be reassessed, consistent with SSG-3 (Rev. 1) [4] paras. 6.17 to 6.19, noting that the latter explicitly states that "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." To determine if such hazards should be taken into account in Level 2 PSA, it should be considered if they can affect the confinement function. In this context, it should be distinguished between:

(a) Hazards, for which the site and plant specific screening has demonstrated that they do not need to be analysed in detail but that a bounding assessment of the Level 1 PSA PDSs (core and/or fuel damage) is sufficient, detailed accident sequences do not have to be modelled, but again a bounding assessment of the radioactive release frequencies (large release frequency or large early release frequency) is sufficient;
(b) Hazards, for which detailed accident sequences have to be modelled and quantified within Level 2 PSA.

5.24. Based on the hazards identification and screening performed within Level 1 PSA, those combined hazards, for which rough or detailed probabilistic analyses have been carried out in the frame of Level 1 PSA, should be analysed within Level 2 PSA. In line with the guidance provided in SSG-3 (Rev. 1) [4] for Level 1 PSA, relevant combined hazards should be treated in the same manner as single hazards when developing accident sequences to be analysed.

## 6. SEVERE ACCIDENT PROGRESSION ANALYSIS

6.1 The severe accident analysis task typically consists of different groups of analyses, performed in different phases of a Level 2 PSA project. Early on in the project, severe accident analysis would be used to understand the general post-core damage accident progression for key initiating events, providing a starting point for the Level 2 PSA analysis (see Ref. [15]). Subsequently, analyses would be performed to support the definition of PDSs, assisting the event tree analysts in establishing which systems and accident progression features are most important for determining the plant response and hence, needing to be included in the PDS definitions. For example, investigations may provide insights on the variation of accident progression when different numbers of injection trains are operating or provide insights into the impact of primary and secondary pressure status (in a pressurized water reactor) or indicate the effect of changes in the volume of water injected to containment on molten core behaviour ex-vessel. A third area where severe accident analyses provide support is in the assessment of specific phenomena, where accident analysis results may be an input to phenomenological probability calculations (see Section 10), and support in defining timing for human action events included in the logic models. Finally, the severe accident analyses support the grouping of accident progression event tree sequences into release categories (see Section 0), a similar process to the definition of PDSs, and are also performed to generate the quantitative characterization of radiological release associated with each release category.

6.2. This section presents recommendations for the Level 2 PSA studies that provide information on the possible severe accident progression paths: accident timing, main phenomena, conditions for manual actions and automatic actions, effects of systems actuation, loads (e.g. pressure, temperature, radiation) induced on the containment boundaries (e.g. building structure, penetrations, seals). The results of these studies are used in containment integrity analysis (see Section 7), in human reliability assessment and equipment reliability assessment (see Section 8), for the development of the accident progression event trees (see Section 9) and can be joined to source term analysis (see Section 10).

6.3. Plant specific deterministic analysis of the progression of accidents inside the reactor should be considered as the preferred method for evaluating severe accident progression and effects in the reactor vessel, in the reactor containment and in auxiliary buildings.

6.4. Severe accident progression analysis should be a significant part of the resources planned to develop a first version of a Level 2 PSA due to the modelling of the plant, the automatic actuations and human actions with appropriate simulation tools.

6.5. Severe accident progression analysis should be performed by teams with experience in application of severe accident codes; if not, training has to be included in the project (see paras. 3.18-3.21 on team selection for the Level 2 PSA project).

6.6. In addition, generic studies of severe accident phenomena and containment response reported in the literature for similar plants could also be used to complement the scope of plant specific calculations to include a broader set of conditions.

6.7. The analysis of the progression of the severe accidents in the reactor, which were identified by the Level 1 PSA and grouped in specific PDSs, should provide key information such as fuel dewatering kinetics, hydrogen production, vessel failure, risk of explosion, risk of basemat penetration by corium, and the amplitude and kinetics of radioactive release.

ANALYSIS OF SEVERE ACCIDENTS INVOLVING REACTOR CORE DAMAGE

6.8. Each identified PDS representing a significant contributor to core damage[17] (see para 5.13 and footnote 15) should be mapped to specific representative calculations, however some calculations can represent more than one PDS, if justified. In addition, calculations could also be performed for those PDSs that may have a low occurrence frequency, but which have the potential to result in large and/or early releases of radionuclides to the environment. Such PDSs typically involve either direct containment bypass or early failure of the primary and/or secondary containment. If detailed calculations are performed for PDSs with high occurrence frequencies and also for those with severe consequences, a sufficiently wide range of information will usually be generated to estimate the response of the plant for other PDSs that are not addressed in detail.

6.9. If relevant, Level 2 PSA should also consider assessment of reactivity accident scenarios resulting in prompt criticality accidents leading to reactor core damage and potential damage to the containment integrity.

[17] A severe accident without core damage may occur if the core geometry is maintained, as might be the case for high temperature gas cooled reactors. In this case, the core damage calculation may be performed in the integral model of Level 1 and Level 2 PSA.

6.10. The analysis of the progression of severe accidents inside the reactor should be performed using one or more computer codes for severe accident simulation (see Annex I for examples of computer codes for water cooled reactors).

6.11. In line with Requirement 18 of GSR Part 4 (Rev. 1) [2], the computer code(s) chosen to perform detailed analyses and the number of calculations that should be performed depend on the objective of the Level 2 PSA. Among the issues that should be considered in making these decisions are:

(a) The code(s) should be capable of modelling most of the initiating events considered in Level 1 PSA, and phenomena that might occur during the progression of the accident according to the state-of-the-art;
(b) Interactions between various physicochemical processes should be correctly addressed in the computer code(s);
(c) The validation and benchmarking effort and the associated documentation should be sufficient to support the necessary severe accident analyses (see for example Ref. [16]);
(d) The code(s) should be verified and validated against the severe accident phenomena treated by them (validation matrix should be available).

6.12. The analysts should be adequately trained in the use of the codes to be applied and be aware of the technical limitations and weaknesses of the selected code(s) (see para. 3.19 on team selection for the Level 2 PSA project).

6.13. The analyses of severe accidents should initially cover key sequences for each PDS leading either to a successful stable state of the plant, where sufficient safety systems have operated correctly so that all the required safety functions necessary to cope with the PDS have been fulfilled, or to filtered containment venting (if provided) or to a degraded state with one or several containment failures. At a second step, remaining sequences in the accident progression event tree should be quantified for confirmation purposes.

6.14. For supporting the Level 2 PSA, deterministic analyses of both the integral behaviour of the plant during a severe accident as well as individual phenomena analyses of the severe accident sequences under consideration are performed. Integral analyses start with the initiating event and end according to appropriate criteria, depending on the purpose of the analysis. Examples of criteria for termination of analyses that have been used are (1) when the cumulative release of radionuclides into environment has stabilized, (2) after corium stabilization (in-vessel or ex-vessel), or (3) after a pre-determined mission time has elapsed. The analysis of individual phenomena should be supported by severe accident analyses as needed by the analysts performing those individual analyses (see also Section 10). Some examples of individual phenomena for water cooled reactors are provided below:

(a) Structural-mechanical behaviour of components of reactor coolant system in the event of high-pressure severe accident scenarios;
(b) Interaction of core, core structures and corium with coolant inside and outside the reactor pressure vessel;
(c) Ex-vessel cooling of the reactor pressure vessel for in-vessel melt retention;
(d) Hydrogen and carbon monoxide generation, flow and distribution in the reactor containment and mitigation means to cope with combustion behaviour;
(e) Ex-vessel corium coolability;
(f) Criticality accidents effects;
(g) Containment pressurization.

6.15. In general, the analyses should be performed in a best-estimate manner regarding applied codes, models, model parameters, as well as boundary conditions. Conservative assumptions for the severe accident analyses, which are common use for the design of nuclear power plants, may not be useful or productive in severe accident analyses for Level 2 PSA because, for example, conservative assumptions may distort the results and risk insights, and consequently may lead to deviation from optimal severe accident management strategies.

6.16. The integral severe accident analyses can also be used for the determination of the source terms (see also Section 9).

6.17. Severe accident management measures for both prevention of core damage as well as mitigation should be considered in the severe accident analyses with realistic timing for human actions.

6.18. All severe accident analyses (assumptions, results) should be part of the Level 2 PSA documentation in order to bring the justification of the probabilistic accident progression (i.e. accident progression event tree) models. Key variables are typically catalogued at important points in time and recorded as time dependent plots for detailed study.

## ANALYSIS OF INTERACTIONS BETWEEN THE REACTOR AND THE SPENT FUEL POOL

6.19. Analysis of the severe accidents in spent fuel pools identified by the Level 1 PSA and grouped in specific PDSs can be performed on the same basis as the analysis of reactor accidents. Section 13 provides guidance on severe accidents in the spent fuel pool.

6.20. Depending on the plant configuration (e.g. whether the spent fuel pool is inside or outside the reactor containment building), severe accident analysis should provide information on the interactions between the reactor and the spent fuel pool: there may be mechanisms whereby a reactor accident can induce a spent fuel pool accident and vice versa. The outcome of this analysis could be some additional accident scenarios (involving both reactor and spent fuel pool) being added to the Level 2 PSA.

6.21. If the spent fuel pool is inside the reactor containment building and the joint accident progression event tree sequences involving the reactor and the spent fuel pool are not demonstrated to be negligible, accident progression analysis should be carried out to show the combined impact of reactor and spent fuel pool accidents on conditions in the containment (e.g. pressure, temperature, corium spreading, inflammable gas, steam production in the containment, acceleration of evaporation of spent fuel pool water) as well as on radiological releases.

## SEVERE ACCIDENT PROGRESSION ANALYSIS FOR LOW POWER AND SHUTDOWN MODES

6.22. Specific analysis should be performed for low power and shutdown modes of reactor operation depending on the reactor technology in order to capture specificities that have implications for the accident progression and for source term calculations. Reference should be made to the plant operational states from Level 1 PSA. The following might apply in low power and shutdown modes:

(a) The core decay heat level might be lower;

(b) It might be possible to open the reactor coolant system with a lower coolant inventory (in this case, high pressure core melt is impossible);
(c) It might be possible to open the containment;
(d) Interconnection between the reactor and the spent fuel pool might be possible (e.g. the possibility to use common safety systems, and common severe accident management guidelines strategies), decay heat loads in each of these locations, and possible fuel assemblies handling.
(e) In water cooled reactors, when the reactor pressure vessel head is closed, provided that the decay heat level and operating configuration are similar to those for full power, core melt accident phenomena could be considered very similar to the sequences that occur in full power mode. When the reactor pressure vessel head is open (i.e. in water cooled reactors), the analyst should consider that some of the Level 2 PSA issues become irrelevant compared to full power mode, while others come into existence. Certain phenomena might not occur (e.g. direct containment heating, induced steam generator ruptures, alpha mode failure). In most shutdown states with an open reactor pressure vessel head, the reactor vessel and spent fuel pool are connected by a large water pool in some reactor designs.

6.23. Specific analysis should be considered for accidents occurring when the reactor and spent fuel pool are connected (see Section 13).

## IDENTIFICATION OF SOURCES OF UNCERTAINTIES

6.24. The analysts should be aware of the technical limitations and weaknesses of the code(s) selected for modelling severe accident progression. The analyst(s) should identify any possible lack of information on plant design or procedures and any lack of information from systems and components qualification, including ageing effects.

6.25. Known areas of uncertainty in the modelling of severe accidents inside water cooled reactors are shown in TABLE 4, together with potential implications on the modelling.

6.26. A plant specific list of uncertain parameters to be varied in the frame of the uncertainty/sensitivity analysis should be derived. The list of parameters for uncertainty analysis should not include parameters such as correlation coefficients, model parameters, etc. used in modelling the phenomenology of severe accidents in the corresponding computer codes, which are established as part of the computer code validation procedure. Otherwise, their variation can lead to completely incorrect results of the uncertainty analysis.

6.27. The uncertainties related to calculated key variables (e.g. peak pressures and temperatures, total mass of corium, mass of combustible hydrogen, timing of major events) should be documented for use in the models for quantification of accident progression (e.g. accident progression event tree).

TABLE 4. EXAMPLES OF AREAS OF UNCERTAINTY RELEVANT TO THE PROGRESSION OF SEVERE ACCIDENTS INSIDE WATER COOLED REACTORS

| Type of severe accident event | Related phenomena |
|---|---|
| In-vessel core degradation | Formation of flow blockages in core |
| | ‘Ballooning’ of cladding and rod failure |
| | Relocation and solidification of molten fuel |
| | Oxidation and hydrogen generation, |

| Type of severe accident event | Related phenomena |
|---|---|
| | Relocation of corium into vessel lower head<br>Corium stratification (metallic/oxidized layers, focussing effect for thermal flux)<br>Ex-vessel cooling (for in-vessel retention) |
| In-vessel forced/natural circulation | Circulation flows in reactor coolant system loops influence by the presence of water in cold leg (direct or counter-current steam flows)<br>Competing mechanisms of degradation and failure of reactor coolant pump seal |
| In-vessel corium–water interactions (Energetic and non-energetic) | Effect of in-vessel water injection (quenching) after recovery of systems/components: pressure peak, hydrogen generation, fuel cooling depending on core degradation progression and water flow rate.<br>Potential for terminating in-vessel fuel degradation<br>Recriticality<br>Steam explosion, high pressure failure of the reactor pressure vessel<br>Releases and transport of radioactive material |
| Failure of primary circuit | Break size<br>Break location |
| Failure mechanisms of reactor pressure vessel and loop boundaries | Melt penetration and cooling within vessel head penetrations<br>Local or global failure of lower head of reactor pressure vessel: mechanical (creep) or melting failure.<br>Impact of ex-vessel cooling<br>Heat-up and creep rupture of reactor coolant system pressure boundary (hot leg nozzle, pressurizer surge line and steam generator tubes). Impact of possible steam generator tubes defaults. |
| High pressure melt ejection and/or direct containment heating | Trapping of melt debris on containment structures<br>Heat release by zirconium oxidation and additional hydrogen production. Debris transport from the vessel/cavity to the containment atmosphere<br>Hydrogen combustion simultaneously with heat transfer to containment atmosphere (pressurization)Releases of radioactive material |
| Ex-vessel corium–water interactions (energetic and non-energetic) | Debris fragmentation and quench (cooling)<br>Quasi-static increase in containment pressure (steam spike)<br>Local dynamic loads to containment from steam explosion (reactor cavity) and possible damage to structures<br>Releases of radioactive material |
| Core–concrete interactions | Erosion of containment structure (basemat) by debris<br>Generation of incondensable and/or combustible gas (such as CO, $CO_2$ and $H_2$)<br>Lateral/axial spreading of debris and potential for contact with containment boundary<br>Corium spreading<br>Corium coolability |

| Type of severe accident event | Related phenomena |
|---|---|
| | Effects of presence of metal within the melt or within the concrete, melt stratification (metallic/oxidized layers)<br>Releases of radioactive material |
| Hydrogen and carbon monoxide combustion | Mixing and/or stratification of flammable gas in containment atmosphere<br>Steam or nitrogen inerting,<br>Hydrogen and carbon monoxide recombination, Ignition (time) and combustion,<br>flame acceleration and transition from deflagration to detonation<br>Heat losses to structures<br>Containment structure response to combustion pressure wave (open doors or blow-out panels, displacement of water pools, etc.)<br>Transport and distribution of combustible gas in secondary buildings and containment venting systems |

## 7. CONTAINMENT INTEGRITY ANALYSIS

7.1. The content of this Section is based on experience with water cooled reactor technologies with containment. It presumes the existence of some type of passive structure with the capability to withstand some of the conditions resulting after severe damage to the reactor core and thus retaining a large portion of the radioactive material. The most common version of such a passive structure in many plant designs is a containment building or a steel containment vessel, which includes associated containment systems. As such, the applicability of this Section depends on the reactor technology and design. In case of non-water cooled reactors with such a containment structure, containment integrity analysis may be conducted depending on the design features and loading conditions on the containment.

7.2. The containment integrity analysis includes both deterministic and probabilistic analysis methods and should include the following:

(a) The capability of the containment to maintain its leaktightness under internal loads (paras 7.4-7.11);
(b) The potential for loss of containment leaktightness due to failure mechanisms induced by severe accident phenomena, such as erosion of concrete structures by direct interaction with molten core debris (i.e. not for in-vessel melt retention) (paras 7.12-7.16 and 9.3-9.6);
(c) The potential for containment isolation failure or containment bypass resulting in a direct leakage pathway to the environment (paras 7.17-7.20 and 9.3-9.6).

7.3. Containment integrity may also be impacted by hazards. For instance, high-level seismic events may directly cause a loss of containment leaktightness. Therefore, a review should be carried out of the impacts caused by hazards included within the PSA scope, to identify any impacts on the containment structure that need to be captured. Usually this is done as part of the Level 1 and Level 2 PSA as discussed in Section 5.

**Containment performance analysis with respect to internal loads**

7.4. In general, the goal of the containment structural analysis with regard to internal loads is to assess a probability of containment failure as a function of pressure and/or temperature under severe accident conditions[18], known as a fragility curve[19] or a fragility (hyper)surface[20]. Typically, material properties vary with containment structure temperature, so pressure-driven failure modes will be affected by temperature conditions. Usually, an enveloping temperature is chosen (based on severe accident analyses) and the overpressure analysis is then carried out assuming those temperature conditions. Development of containment fragility curves should consider pressure-driven and temperature-driven failure modes which are applicable to the containment design under consideration in the Level 2 PSA. This analysis should include an identification of the potential failure mode(s) and their respective location(s).

7.5. A realistic characterization of the leakage area associated with each failure mode should be developed. Failure criteria are also needed for each failure mode. Typically, in terms of leakage area, a classification is made into small area failures (usually designated as 'leaks') and large area failures (usually designated as 'ruptures' and/or 'catastrophic ruptures'). Criteria for containment leakage allowed (e.g. threshold expressed as volume percentage per day) or containment failure under severe accident conditions (e.g. a realistic allowed structural deformation) should be defined. Design criteria for the containment are generally not adequate best-estimate measures of capacity of the containment because of the safety factor built into such values. Furthermore, containment design limits may not account for the harsh environmental conditions that can develop inside the containment during a severe accident, for which additional failure modes often need to be considered.

7.6. To generate a realistic assessment of containment performance limits, detailed information on the structural design of the containment and containment penetrations (see TABLE 5 providing a high level list) should be collected. In the collection of information for the analysis, particular consideration should be given to the potential for leakage through a steel liner (if any) or penetrations.

TABLE 5. EXAMPLES OF IMPORTANT FEATURES OF THE STRUCTURAL DESIGN OF THE CONTAINMENT AND CONTAINMENT PENETRATIONS FOR WATER COOLED REACTORS

| | |
|---|---|
| Containment type | Steel<br>Concrete:<br>— Prestressed<br>— Post-tensioned |

[18] Examples of phenomena leading to overpressurisation loads could be combustion of combustible gases such as hydrogen combustion (i.e. deflagration and detonation) and carbon monoxide combustion.
[19] Fragility curve representing the probability of containment failure as a function of one variable, such as pressure or temperature.
[20] Fragility surface representing the probability of containment failure as a function of more than one variable together, such as pressure and temperature.

| | |
|---|---|
| | — Reinforced<br>— Presence or absence of steel liner<br>— Presence or absence of resin (e.g. epoxy) liner |
| Containment penetrations | Equipment hatch(es)<br>Personnel hatch(es)<br>Piping penetrations<br>Electrical penetrations<br>Atmosphere purge line(s)<br>Vent line(s) |
| Other aspects | Geometrical shape of containment (sphere, cylinder, rectilinear)<br>Geometric details of the containment structure, penetration and hatches<br>Geometrical discontinuities (e.g. transition from cylindrical shell to top head and basemat)<br>Steel or resin (e.g. epoxy) liner anchorages<br>Details of any reinforcements around penetrations<br>Materials used<br>Interactions with other surrounding structures |

7.7. This step of the Level 2 PSA is aimed at developing a best-estimate assessment of the ultimate strength of the containment. This can be done by completing plant-specific structural calculations which account for the containment design features shown in TABLE 5. However, depending on the scope of the Level 2 PSA, use can be made of existing calculations for plants having similar containment designs. In this case, the PSA documentation should provide a thorough justification for the use of existing calculations. Items to address in this justification include similarities and differences of the designs, applicability of the existing structural response analyses to the plant under consideration, and basis for any adjustments or extrapolations made.

7.8. When applying the fragility curves in the accident progression event tree models, calculations should be made of the probability of containment failure for each of the different leakage area categories defined according to paragraph 7.5, to the extent that these separate calculations are necessary to realistically address the objectives of the Level 2 PSA. In calculating the probability of a rupture failure mode, credit may be taken for two basic approaches used in PSA studies to characterize the loss of containment integrity, namely the ‘threshold model’ and the ‘leak before break model’. The threshold model defines a threshold pressure, with some associated uncertainties, at which the containment is expected to fail. This failure is represented as a large rupture and with the potential for a significant and rapid blowdown from the containment atmosphere to the environment. In the leak before break model, containment leakage failure modes occur at pressures below the pressure at which a larger failure mode occurs. The precise treatment depends on the rate of addition of mass and energy to the containment. If this rate of addition is smaller than or equal to the leakage rate associated with a leak failure, containment pressurization would be stopped once such a failure occurs, thus preventing the larger rupture failure mode from occurring.

7.9. If plant specific calculations are necessary, containment performance analyses should be based on validated structural models supported by data and reasonable failure criteria. In particular, the failure criteria used in plant specific calculations should be justified. The use of containment failure experiments may be useful for this purpose (see para. 7.24). In the analysis, fragility curves should be developed for static pressure loads. A best practice would be to include pressure ramp rates, localized heat loads and localized or global dynamic pressure loads. Dynamic pressure loads could be addressed in a simplified way, for example by a single degree of freedom calculation. Ageing of structures should also be taken into account. The supporting analyses provide an engineering basis for containment failure mode, location, size, and ultimate pressure and/or temperature capabilities of the containment structure.

7.10. Large penetrations (e.g. material access, personnel access) and singularities zones can be a relative weak point of the reactor building in severe accident conditions. The impact of these penetrations on containment capacity should be considered in the reactor building structural analysis. This may include performing global analysis followed by local analysis for these zones with adequate detail to capture the local mechanical behaviours. Usually such local analyses take boundary conditions from the global containment structural model. As an example, the behaviour of the material access closure system should be studied under severe accident conditions (see Refs [17], [18]).

7.11. While internal pressure loading is the principal determinant of potential containment failure, consideration should also be given in the Level 2 PSA to the possible effects of temperature and radiation on the containment performance. The temperature of the containment could affect the strength characteristics of the structural materials (see Ref. [18]) as well as cause degradation of penetration seal materials. The impact of radiation on penetration seal or gasket materials could also affect the tightness properties of the seals in a severe accident environment, in particular if they are directly exposed to radiation.

**Analysis of containment leaktightness due to other failure mechanisms induced by severe accident phenomena**

7.12. Containment leaktightness might be also affected by failure mechanisms induced by severe accident phenomena. Examples of phenomena to consider could be induced fires (e.g. graphite fires), steam explosion (e.g. instantaneous vaporization of water induced by its contact with molten corium), chemical attack (e.g. chemical reactions affecting containment structures integrity) and direct contact between molten core debris and containment structures. Recommendations related to the consequences of molten core debris and containment structures for the containment integrity analysis are presented in paras 7.13 to 7.16.

7.13. The effects of extensive erosion of concrete structures due to long term exposure and to attack by molten core debris (molten core–concrete interactions) should be examined if calculations of severe accident progression suggest extensive erosion due to such interactions are possible (see Section 6). This may be of particular concern for the response of the containment basemat but also, according to the plant design, the response of containment wall, or reactor pressure vessel support structure (e.g. concrete pedestal).

7.14. Extensive erosion of concrete structures may lead to containment leakage or failure if the structures interacting with the molten core debris are significantly weakened due to this interaction. Failure criteria for these mechanisms, which is generally expressed in terms of reduction of concrete thickness, should be defined and justified.

7.15. The consequences of an extensive erosion of concrete structures should be examined. For example, the response of a reactor pressure vessel support structure (e.g. concrete pedestal), containment wall or floor to the complete or partial penetration by core debris.

7.16. Potential locations for melt through of the containment (e.g. penetrations, sump suction lines) should be identified and analysed.

**Analysis of initial and induced containment isolation failure and containment bypass**

7.17. The potential for containment isolation failure should be assessed. All the containment penetrations should be modelled or a careful justification has to be provided to justify the screen out of some penetrations. Screening criteria may be applied in order to focus on the relevant penetrations that are most likely to result in important releases. For instance, containment isolation may not be modelled for normally closed lines provided that isolation valves would not be opened during the accident (e.g. due to the initiating event or type-A human failure event) or for closed loop systems inside the containment provided that closed loop integrity will not be threatened during the accident. If any, the plant operating feedback regarding containment isolation valves leakages should be taken into account.

7.18. Possible failure modes of valves should be taken into account: failure to close, spurious opening; leak through closed valves may also be considered depending on the goals of the Level 2 PSA (as low levels of leakage are not likely to cause large releases). The dependencies involved in containment isolation should also be taken into account (e.g. power supply to motor valves, automatic isolation signals). In addition, manual recovery operator actions can be credited if justified.

7.19. The potential for containment bypass (release from the core to the environment without being able to credit containment) should be assessed if not already provided through interface with the Level 1 PSA. The bypass paths should be identified by a rigorous search of systems located outside the containment and linked to reactor coolant loops. In general, containment bypass may result either from an initiating event (e.g. steam generator tube rupture or loss of coolant accidents of interfacing system), or from an induced event due to accidental conditions (e.g. severe accident induced steam generator tube rupture for a pressurized water reactor), or from a failure (leak or an incorrect circuit configuration, etc.) in an emergency cooling line outside of the containment.

7.20. Potential containment failure modes due to isolation failure or bypass that have not already be taken into account in the Level 1 PSA should be addressed in the Level 2 PSA.

## CONTAINMENT INTEGRITY ANALYSIS FOR LOW POWER AND SHUTDOWN MODES

7.21. For low power and shutdown modes of operation, the primary considerations relevant to containment integrity analysis are associated with the potential for large containment penetrations to be present such as an opened equipment hatch. If this condition is present, the assessment of containment internal loading and the effects on concrete structures due to molten core debris interactions is not applicable to the assessment of containment integrity since the containment is in a bypassed state. However, the degradation of structures due to interactions with core debris may still be relevant to other parts of the Level 2 PSA. If the containment remains sealed during low power and shutdown modes of operation, then the previous recommendations within this chapter are applicable.

**Analysis of containment isolation failure during shutdown**

7.22. With regard to containment isolation failure, particular attention should be paid to shutdown modes of operation for which:

(a) Personnel airlocks or equipment hatch may be open at the time of the initiating event. The closure of such very large penetrations before the onset of core damage should be precisely evaluated. In particular for the equipment hatch, the analyst should take into account if the closure is requested by emergency operating procedures, the delay needed to perform the closure, the dependencies involved (power supply), etc.
(b) Isolation lines normally closed at power that may be opened and may not be subject to automatic isolation.

CHARACTERIZATION OF UNCERTAINTIES

**Characterization of uncertainties related to containment performance under internal loads**

7.23. In determining the structural performance of the containment, the uncertainties associated with estimation of the structural capacities necessary for withstanding extremes of pressure and/or temperature should be assessed. Uncertainties arise in the evaluation of the ultimate strength capacity as the result of several factors including:

(a) Material variability, which characterizes uncertainties about intrinsic properties of the materials, such as the behaviour laws, yield and tensile strengths, influence of the temperature on the mechanical characteristics, etc.
(b) Modelling uncertainty, which characterizes uncertainties about the geometry of the model (e.g. position and section of the materials), the material failure models considered, or the reliability of calculations performed.

7.24. Material variability and modelling uncertainties can be determined by techniques for uncertainty quantification and propagation, as part of the structural capacity assessment in order to establish a failure pressure and/or temperature distribution function. Statistical feedback based on test samples from construction sites may be useful to assess material variabilities. Benchmark from mock-ups or feedback experience from pressure tests (if available) may be useful to assess modelling uncertainty. Model uncertainty may be assessed via reference to Sandia large scale simulated containment failure experiments (see Ref. [19]). These tests include pressure tests on 1:8 scale steel pressure vessels and 1:6 scale reinforced concrete containments and variously scaled prestressed concrete containments. Experiments are supported by analytical predictions of containment failure pressure; these analytical predictions are useful to transpose knowledge at the reactor scale (see e.g. Ref. [17]). Alternatively, expert judgement supported by simple analysis could also be used to establish this failure pressure and/or temperature distribution for various credible failure modes (leaks and ruptures).

7.25. Containment failure is typically described by a composite fragility curve calculated by combining the individual containment failure mode fragility curve. Note that the individual curves are also needed for calculation of leak and rupture failures as described in para 7.8. These individual fragility curves may also be applied in the accident progression event tree task when competing leak and rupture failure modes are evaluated. Each fragility curve should be characterized by a best estimate (median) failure pressure, a parameter representing the material variability and a parameter representing the modelling uncertainty (see para 0).

**Characterization of uncertainties related to concrete structures erosion by molten core debris**

7.26. Uncertainties associated with concrete structures resistance toward extensive erosion by molten core debris should be assessed. In a similar way as above, uncertainties are the result of several factors including material variability and modelling uncertainty.

7.27. Relevant uncertainties affecting the development of the molten core–concrete interactions phenomenon include the availability of water (presence before vessel failure or injection after vessel failure), containment geometry, corium temperature, amount and composition of core debris, decay power and the type of concrete used for the basemat construction.

7.28. The molten core–concrete interactions phenomenology is rather complex and various situations may occur as the result of the accident progression. Assessment of the probability of an extensive erosion of structures should account for the uncertainties affecting the molten core–concrete interactions calculations.

7.29. Those uncertainties can be assessed, for example by using the OECD NEA state-of-the-art report regarding molten core–concrete interactions [20] in which various topics related to these interactions, such as available experiments, plant application, simulation tools and models, and uncertainties are comprehensively summarized.

**Characterization of uncertainties related to containment isolation failure and containment bypass**

7.30. The uncertainties associated with estimation of containment isolation failure and containment bypass should be assessed. This may include uncertainties associated with the following items:

(a) Data used (component reliability data, maintenance unavailabilities, opening duration of large penetrations such as equipment hatch or personnel airlocks);
(b) Human reliability assessment (see Section 8);
(c) Severe accident phenomena modelling for induced containment bypass (see Section 6).

# 8. HUMAN AND EQUIPMENT RELIABILITY ASSESSMENT

## HUMAN RELIABILITY ASSESSMENT

8.1. Human failure events can be classified in the same way for Level 2 PSA as for Level 1 PSA (seeSSG-3 (Rev. 1) [4]):

(a) Type A human failure events are those that occur before the initiating event, that have the potential to lead to the failure or unavailability of SSCs important to safety. Level 2 PSA may include Type A human failure events associated with the systems not considered in Level 1 PSA;
(b) Type B human failure events are those that cause an initiating event. These events are

included in the Level 1 PSA, but not relevant to Level 2 PSA;

(c) Type C human failure events are those that respond to an accident (initiating event). Identifying and analysing Type C human failure events is the main task performed in human reliability assessment for Level 2 PSA. Paragraphs 8.2-8.12 deal with Type C human failure events in Level 2 PSA. Examples of Type C human failure events include actions to arrest and mitigate a core damage condition, protect containment from failure, and terminate or mitigate fission product releases.

8.2. Human actions that impact the severe accident progression or the releases of radionuclides should be modelled in Level 2 PSA. In general, these actions are identified in:

(a) Emergency operating procedures, specifically the actions taken under these procedures that are not modelled in Level 1 PSA but are relevant in Level 2 PSA (e.g. manual containment isolation, containment isolation in shutdown mode, reactor coolant system depressurization, application of entry criteria for entering severe accident management guidelines);
(b) Severe accident management guidelines, including:
   (i) Operator actions that can be implemented without any crisis teams support and/or approval;
   (ii) Operator actions that can be implemented with the support and decision of the crisis organization (or technical support centre);
(c) Strategies and guidelines for deployment of non-permanent equipment or additional strategies not considered in emergency operating procedures, if such strategies and guidelines have been implemented.

8.3. These actions are most often carried out by operators (in the control room), or by field workers (outside the control room). However, some of them can also be carried out by external teams (i.e. the team external to the plant organization) specially trained to handle severe accidents (for example, an external response organization that transports non-permanent equipment staged in a distant location to the site to mitigate the event).

8.4. Depending on the objectives and intended uses with Level 1 PSA, it is advised to revise the PSA level 1 human reliability assessment to reassess level 1 operator actions from a Level 2 PSA perspective (e.g. conservatism may have been used resulting in too high numbers).

8.5. Human actions (e.g. manual opening of pressure operated relief valves) needed before or soon after the onset of core damage can be represented in the extended accident sequence event trees in the Level 1 PSA model if it can be justified that the performance of the action is feasible. In such cases, the status of such human actions (success or failure) should be reflected either explicitly by an attribute of a PDS or implicitly via their impacts on the status of the attributes already defined for the PDS. Other relevant severe accident management actions that are not represented in Level 1 PSA should be incorporated into the Level 2 PSA. Typically, such actions are those expected to occur later in the severe accident sequence, for example, refilling steam generators to reduce the releases to the environment via damaged steam generator tubes, restarting the low pressure injection after a high temperature induced break in primary circuit boundaries (primary cooling system), or opening the containment venting line to relieve containment pressure.

8.6. Human reliability assessment for Level 2 PSA should be consistent with Level 1 PSA, for which recommendations are given in SSG-3 (Rev. 1) [4]. In particular, the effects induced by hazards or hazards combinations on the performance shaping factors of operating personnel should be taken into account in a coherent way between Level 1 PSA and Level 2 PSA.

8.7. However, specificities in the human reliability assessment for a Level 2 PSA should be taken into account, such as:

(a) How human actions are prescribed. Depending on the organization put in place to deal with a severe accident, some actions may be carried out independently by the plant staff, while the others need to be approved or directed by the crisis organization. In the latter case, this requires the crisis organization to be fully operational and a good coordination with plant staff;
(b) The instructions in severe accident management guidelines may be less specific than emergency operating procedures. The lack of specifics may increase the likelihood of human error, including errors of commission and errors of omission;
(c) How human actions are performed. The potential impact of degraded workplace conditions (in particular high radiation and/or high temperature in certain rooms, lighting potentially degraded or absent) should be taken into account, especially when the necessary action is carried out locally, because these conditions may affect human reliability.

8.8. APPENDIX I provides more detailed information about performing human reliability analysis for a Level 2 PSA. Human reliability assessment in the context of a Level 2 PSA for multiple unit nuclear power plants is covered in Section 14.

8.9. Assessment of human reliability in the context of deploying non-permanent equipment should follow the same principles as in Level 1 PSA human reliability assessment. Assessment should consider in particular:

(a) Adverse conditions on site and nearby (e.g. climatic conditions, road access, radiological conditions due to progression of severe accident);
(b) Delay in actions because of simultaneous actions that share the same resources (e.g. component, water, and manpower) or the other actions having higher priority;
(c) The existence of site specific procedures to implement non-permanent equipment and the realization of regular exercises on site;
(d) If the non-permanent equipment is provided and implemented by external teams, the coordination between the plant organization and the external teams.

8.10. It is important to ensure that potential dependencies between operator actions should be assessed and taken into account when appropriate. This includes the dependencies between the human actions credited in Level 2 PSA and the dependencies between the human actions credited in Level 1 PSA and Level 2 PSA, noting that strong dependency can occur if the human actions are performed by the same operators, if they involve the same equipment, or if the actions are close in time.

8.11. Potential adverse effects of severe accident management actions should be considered (e.g. as part of the event tree logic). For instance, injecting water into a degraded core may arrest the progression of a severe accident, but with potential side effects, including energetic fuel–coolant interaction, fuel shattering and additional release of steam, hydrogen and radioactive material. The potential phenomena and their effects on scenario and human reliability should be evaluated.

8.12. The results of the Level 2 PSA can, and should, be used to identify or improve severe accident management actions as explained in Section 15.

EQUIPMENT RELIABILITY ASSESSMENT

8.13. Equipment reliability in a Level 2 PSA is usually modelled using the same techniques as applied in the Level 1 PSA – for example, data analysis and fault tree construction. Reliability models such as fault tree models are usually linked into the accident progression event tree models or included by the extended Level 1 event trees or the bridge trees. Recommendations related to equipment reliability assessment in Level 2 PSA models are presented in paras 8.14 to 8.17.

8.14. Assessment of the reliability of equipment credited within the Level 2 PSA should consider the periodic testing and maintenance practices or planned procedures. Such practices or procedures may differ from those used for the systems and components credited within Level 1 PSA and thus may have an influence on systems reliability.

8.15. The effect of the environmental conditions resulting from a severe accident on the survivability of components and systems credited within the Level 2 PSA model should be assessed. Some components and systems may already be qualified to severe accident conditions. Otherwise (or if severe accident conditions exceed qualification profiles), the survivability assessment should be based on supporting studies or expert judgment. Adverse environmental impacts may include containment/auxiliary buildings high temperature, pressure, humidity and radiation conditions. Examples of adverse conditions that could affect equipment reliability are energetic events (e.g. short term temperature and pressure spikes or impulse loadings from detonations or steam explosions) or high radiation environment (e.g. the electronic instrumentation, rubber gaskets that could be vulnerable to high radiation).

8.16. Repair actions should be credited in Level 2 PSA only if there is strong justification for their feasibility. It might be possible to credit repair actions if the specific failure mode of the equipment is known for the specific sequence and (i) it is possible to diagnose the failure, (ii) the spare parts and repairing personnel are in place, (iii) the environmental and work conditions needed for performing repair are given or they can be ensured, and (iv) the time window is sufficiently long to credibly assume the possibility for repair, including the time needed to bring spare part and repairing personnel to the plant. Reliable data should moreover be used to assess credible probabilities of repairing components and systems. For the components that are not reparable after a severe accident occurrence and that are continuously required after core melt (for corium cooling, for example), their failure probability assessment should integrate this long mission time. A discretization of the failure modelling for different time windows could be implemented to consider different consequences as a function of the instant of failure.

8.17. Dependencies relating to system availability should also be correctly taken into account between Level 1 PSA and Level 2 PSA.

IDENTIFICATION OF SOURCES OF UNCERTAINTIES IN RELIABILITY ASSESSMENT

**Human reliability assessment**

8.18. Uncertainties about human reliability assessment should be addressed in the same way as for human reliability assessment credited in Level 1 PSA.

8.19. The analyst should assess human reliability uncertainty based on the uncertainty of the factors affecting the human reliability. The factors include but not limited to the duration of the response time window, duration of the human action, environmental conditions, quality of procedural guidance, operator training, and the coordination between the plant staff organization and crisis organization after entering severe accident management guidelines.

8.20. Many human reliability assessment methods use a simplified approach to assess uncertainties by providing an error factor applicable to different human error probabilities. Sensitivity analyses should be performed to evaluate the range of human reliability affected by the key factors.

**Equipment reliability assessment**

8.21. Uncertainties about equipment reliability data should be addressed in the same way as for the equipment credited in Level 1 PSA.

8.22. Uncertainties about equipment qualification or survivability with regard to severe accident conditions should be addressed taking into account areas of uncertainty both related to the evaluation of specific Level 2 PSA environmental conditions and to the resilience of the equipment.

## 9. DEVELOPMENT OF ACCIDENT PROGRESSION EVENT TREES AND QUANTIFICATION OF EVENTS

DEVELOPMENT OF ACCIDENT PROGRESSION EVENT TREES

9.1. For the development of a Level 2 PSA event tree model, two different approaches can be used: an integrated approach and a separated approach as described in para 2.6 The Level 2 analyst should be well trained and aware of the limitations and the requirements imposed by the use of the approach and the computer codes chosen. More information can be found in Ref. [21] from the ASAMPSA2 project.

9.2. In Level 2 PSAs, event trees are used to delineate the sequence of events and severe accident phenomena after the onset of core damage that challenge containment integrity and the successive barriers to radioactive material release. They provide a structured approach for the systematic evaluation of the capability of a plant to cope with severe accidents. Their use is shown in FIG. 1. Such event trees, termed accident progression event trees in this safety guide, include modelling of phenomena, systems actuation or failure, human actions and all impacts on the confinement of radioactive products or the radioactive releases in the environment.

STRUCTURE OF ACCIDENT PROGRESSION EVENT TREES AND NODAL QUESTIONS

9.3. The nodal questions (also referred to as top events) in an accident progression event tree[21] should address the events and physical processes that govern accident chronology, plant response to severe accident conditions, the success and failure of SSCs and human provisions intended for severe accident, relevant challenges to containment integrity and associated barriers to radioactive material release, the physical containment conditions at the time of radioactive material release, and the eventual magnitude of the release of radioactive material to the environment. The nodal questions of the accident progression event tree are specific to plant type, i.e. issues of importance to severe accident behaviour in one type of reactor and/or containment system may not be important to others. The complexity of accident progression event tree depends on the scope and objectives of the Level 2 PSA.

9.4. The list of such events and processes can be rather extensive. Therefore, accident progression event trees can grow to become rather large with complicated logic models. However, relatively simple logic models can be sufficient for certain applications. Thus, for instance, if the objective of the Level 2 PSA is solely to determine the large early release frequency and a quantitative assessment of the full range of severe accident source terms is not necessary, smaller accident progression event tree structures can be developed that focus on severe accident sequences with high consequences within the appropriate time frame[22] (see Ref. [23]). In any case, the overall structure of the model should be traceable by independent reviewers and manageable by the PSA team. Therefore, in the accident progression event tree structures, a reasonable balance between modelling detail and practical size should be achieved.

9.5. The accident progression event tree structure should be phenomenologically and chronologically consistent, should properly take into account interdependencies among events and/or phenomena depending on the reactor technology and should reflect an appropriate level of detail to satisfy the objectives of the Level 2 PSA. Regarding chronology, it is both useful and common practice to divide the accident progression event tree into phases sequential in time, with the transitions between phases representing important changes in the issues that govern accident progression. Examples of phases for water cooled reactors are:

(a) Phase 1: Immediate response of the plant to the PDS caused by the initiating event through the early period of in-vessel core damage;
(b) Phase 2: Late period of in-vessel core damage up to failure of the reactor pressure vessel;
(c) Phase 3: Reactor pressure vessel rupture and its consequences;
(d) Phase 4a: Short-term ex-vessel phenomena and events;
(e) Phase 4b: Long-term ex-vessel phenomena and events.

9.6. Phase 3 is close to the time of reactor pressure vessel failure (to address challenges occurring due to failure of the reactor pressure vessel, e.g. direct containment heating). Phase 4 covers both the short and the long term ex-vessel phenomena and events after the reactor

[21] Nodal questions also address issues and actions relating to severe accident management.
[22] The ASME/ANS Level 1 PSA Standard [23] includes requirements for developing event trees capable of assessing large early release frequency. In the United States the resulting large early release frequency metric is used in regulatory submittals.

pressure vessel rupture. Phase 4a is up to a few hours after failure of the reactor pressure vessel (to address immediate ex-vessel molten core behaviour, e.g. stabilization of the melt ex-vessel or onset of the core–concrete interaction, human actions and equipment behaviour). Phase 4b is the long term, starting from a few hours after failure of the reactor pressure vessel (to address challenges arising from ex-vessel melt behaviour, e.g. pressurization due to the generation of non-condensable gases during core–concrete interaction or combustion phenomena or pressurization due to ongoing steam generation, human actions and equipment behaviour) (see paras 7-7.22). Typical nodal questions for accident progression event tree used in separated models for a typical pressurized water reactor with a large, dry containment are provided in TABLE 6. These or similar nodal questions should also form a basis for the accident progression event tree when the integrated model is used. These are only examples: in practice, nodal questions and their prior dependencies should be precisely developed by the analysts in accordance with the plant-specific reactor technology, design and severe accident management strategies.

TABLE 6. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR

| | Nodal question | Dependencies | Technical bases |
|---|---|---|---|
| Phase 1: Initiating event through early period of in-vessel core damage | | | |
| 0 | Is the accident induced by a core prompt criticality with immediate consequences for the vessel or the reactor containment? | None | Based on PDS and accident progression studies |
| 1 | Is the containment isolated? | None | Based on PDS |
| 2 | What is the fraction of the PDS with alternating current power available? | None | Based on PDS |
| 3 | What is the mechanical status of sprays in the very early time frame? | None | Based on PDS |
| 4 | What is the mechanical status of fans in the very early time frame? | None | Based on PDS |
| 5 | Is the reactor coolant system depressurized manually in the very early time frame? | 2 | Based on emergency operating procedures |
| 6 | Does a temperature induced ‘hot leg’ failure occur in the very early time frame? | 5 | Accident progression |
| 7 | Does a temperature induced rupture of a steam generator tube occur in the very early time frame? | 5, 6 | Accident progression |
| 8 | Is alternating current power restored or maintained in the very early time frame? | 2 | Based on PDS |
| 9 | Are containment sprays actuated in the very early time frame? | 3, 6, 8 | Accident progression |

| | Nodal question | Dependencies | Technical bases |
|---|---|---|---|
| 10 | Does hydrogen combustion occur in the very early time frame and what is the induced pressure peak in the containment? (does it impact (1) the FP releases (resuspension), (2) the containment integrity, (3) specific equipment in the containment (local effects)) | 4, 5, 6, 8, 9 | Accident progression |
| 11 | Does the containment fail in the very early time frame? | 1, 10 | Accident progression |
| 12 | Is containment isolation recovered in the very early time frame? | 1, 8 | Based on PDS |
| 13 | Is the containment filtered vent system actuated in the very early time frame? | 1, 10, 11 | Accident progression |
| Phase 2: Late period of damage progression, including failure of the reactor pressure vessel | | | |
| 14 | Is core damage arrested in the vessel, preventing a failure of the reactor pressure vessel? | 5, 6, 7, 8 | Design features of the reactor pressure vessel and Accident progression |
| 15 | Does an energetic fuel–coolant interaction occur and breach the reactor pressure vessel and containment? | 5, 6, 7, 14 | Accident progression |
| 16 | What are the mode of reactor pressure vessel failure and the process of core debris ejection? | 5, 6, 7, 14, 15 | Accident progression |
| 17 | Does 'rocketing' of the reactor pressure vessel occur and breach the containment? | 16 | Accident progression |
| Phase 3: Reactor pressure vessel rupture and its consequences | | | |
| 18 | Is the under-vessel region flooded or dry at breach of the reactor pressure vessel? | None | PDS and design |
| 19 | What is the mode of under-vessel fuel–coolant interaction following breach of the reactor pressure vessel? | 16, 18 | Accident progression |
| 20 | Does hydrogen combustion and direct containment heating occur at breach of the reactor pressure vessel? | 4, 8, 9, 10 14, 16 | Accident progression |
| 21 | Does the containment fail at failure of the reactor pressure vessel? | 1, 11, 13, 15, 16, 19, 20 | Accident progression |
| 22 | Does the filtered vent system actuate at breach of the reactor pressure vessel? | 1, 11, 13, 15, 16, 19, 20, 21 | Accident progression |

| | Nodal question | Dependencies | Technical bases |
|---|---|---|---|
| Phase 4a: Short-term ex-vessel phenomena and events | | | |
| 23 | Is alternating current power restored or maintained in the short time frame? | 8 | Based on PDS |
| 24 | Do sprays actuate or restored to operate in the short time frame? | 23, 9 | PDS/accident progression |
| 25 | Do fan coolers actuate or restored to operate in the short time frame? | 4, 23 | Based on PDS |
| 26 | Is core debris in a coolable configuration outside the vessel? | 16, 18, 19, 15, 17 | Design features of the core catcher, or accident progression |
| 27 | Is a containment heat removal system in operation or restored during the short time frame? | 1, 10, 23, 24, 26 | Accident progression |
| Phase 4b: Long-term ex-vessel phenomena and events | | | |
| 28 | Is alternating current power restored or maintained in the late time frame? | 23 | Based on PDS |
| 29 | Do sprays actuate or continue to operate in the late time frame? | 24, 28 | PDS/accident progression |
| 30 | Do fan coolers actuate or continue to operate in the late time frame? | 25, 28 | Based on PDS |
| 31 | What is the status of fans and containment sprays in the late time frame? | 29, 30 | Summary type question |
| 32 | Does hydrogen combustion occur in the late time frame and what is the induced pressure peak in the containment? (does it impact (1) the fission product releases (resuspension), (2) the containment integrity, (3) specific equipment in the containment (local effects)) | 10, 20, 31 | Accident progression |
| 33 | Does the filter vent system actuate in the late time frame? | 1, 10, 11, 13, 15, 19, 20, 21, 26, 28, 31, 32 | Accident progression |
| 34 | Is a containment heat removal system in operation during the late time frame ? | 1, 10, 28, 29, 32 | Accident progression |
| 35 | Is the integrity of the containment basemat maintained? | 11, 12, 21, 22, 26, 33, 34 | Accident progression |

| | Nodal question | Dependencies | Technical bases |
|---|---|---|---|
| 36 | Does containment failure occur in the late time frame (slow overpressurization, hydrogen combustion)? | 1, 10, 11, 13, 15, 19, 20, 21, 26, 32 | Accident progression |
| 37 | What are the modes of containment failure? | 11, 21, 35 | Accident progression |

QUANTIFICATION OF EVENTS

9.7. The assignment of conditional probabilities to branches of the accident progression event tree (or other modelling associated with a nodal question) should be supported by documented analyses and data to provide a justified representation of the uncertainty in the outcome at each node. Account should be taken of issues that could affect the analyst's ability to predict the progression of severe accidents and assigning uncertainties, including limitations in knowledge regarding aspects of severe accident phenomena, model completeness, fidelity and validation of available computer codes, applicability of available experimental data to full scale reactor conditions, etc. Example methods for dealing with such uncertainties and the use of expert judgement and expert elicitations can be found in Refs [24]–[31].

9.8. The rationale used to develop appropriate probabilities for each branch can sometimes be made more traceable by decomposing the problem into a number of sub-issues according to the governing phenomena [32], [33]. Such assessments may be carried out separately and reported in support documentation of the results that are used in the nodal questions of the accident progression event tree or may be an integral part of the accident progression event tree in the form of decomposition event trees that are linked to the headings of the accident progression event tree. The degree to which the assessments are integrated into the quantification of the accident progression event tree is principally dependent on the capabilities of the software being used for quantification of the Level 2 PSA. Linked event trees, fault trees (see e.g. Ref. [34]), user defined functions and other methods have been used for developing and quantifying accident progression event trees (see Ref. [35]) .

9.9. Regardless of the approach taken to establish values for the probabilities of events, the process should be traceable so that others can follow and understand the technical rationale, and it should be applied consistently to the full range of events or questions described in the accident progression event tree. Level 2 PSA model usually involves events of different natures: system operation, human action, containment response or components response to severe accident phenomena. Recommendations for assessing human action failure and equipment failure in a context of a Level 2 PSA are provided in Section 8. Paragraphs 9.10 to 9.15 provide recommendations on determining probabilities associated with containment or components response to severe accident phenomena.

9.10. Sources of current and relevant information should be used to support the assignment of probabilities. Information used to support quantification of probabilities can include:

(a) Basic principles or phenomenological-specific models for treatment of relevant severe accident challenges;
(b) Results and/or insights of supporting deterministic analyses using established computer codes for modelling severe accidents;
(c) Relevant experimental measurements or observations;

(d) Results and/or insights of analyses and findings from studies of similar plants;
(e) Expert elicitation involving independent experts.

9.11. Several methods and tools are available to translate such information into a numerical value for each probability. Two simple tools, the threshold approach and the convolution approach, are briefly described in this Safety Guide. NUREG-1150, Reference [22], has historically been a key source of information for many Level 2 PSAs. However, the state of knowledge of severe accident phenomena has progressed since the NUREG-1150, Ref. [22] study, thus reducing its usefulness as a reference for modern Level 2 PSA studies, which should reflect the current state of knowledge. A compilation of recent, relevant severe accident phenomena can be found in Refs [36], [37], as well as work associated with Advanced Safety Assessment Methodologies: Level 2 Probabilistic Safety Assessment (ASAMPSA2) project [21]. Developments have taken place in a number of areas for water cooled reactors, such as:

(a) In-vessel steam explosions (alpha mode containment failure), e.g. Ref. [32];
(b) Direct containment heating, e.g. Refs. [38], [39];
(c) Failure of the lower head of the reactor pressure vessel, e.g. Refs [40], [41];
(d) Flame acceleration and the transition from deflagration to detonation, e.g. Ref. [42];
(e) Thermally induced steam generator tube rupture and hot leg failure Ref. [43];
(f) Recovery of partially degraded cores Refs. [44], [45].

9.12. Experimental programmes regarding the response of containments to internal pressurization conditions beyond design basis that may be useful in supporting development of containment fragility models is provided in Ref. [19].

**Threshold approach**

9.13. The threshold approach can be used to estimate the probabilities of events that occur when the predicted accident conditions approach an established limit or criterion. The failure probability is, therefore, a function of how close the parameter is to the failure threshold. The assignment of numerical values is thus indicative of the analyst's confidence in the rigour, applicability and completeness of deterministic predictions of relevant phenomena.

**Convolution approach**

9.14. In the convolution approach, a higher degree of mathematical rigour is applied to the comparison of how close the parameter of interest (e.g. pressure, temperature) is to the failure threshold (e.g. failure pressure, failure temperature). Both the parameter of interest and the failure threshold are treated as uncertain parameters. Probability density functions representing probability distributions of uncertain parameters are arrived at on the basis of deterministic analyses and expert judgement, and the overlap and/or interference of two such probability distributions determines the degree of 'belief' in (the subjective probability for) failure. In this case, the consistency of the resulting probability values is dependent on consistent assignment of distribution parameters (median values, deviations about the median, choice of distribution type and limits).

9.15. Both approaches, the threshold approach and the convolution approach, can be applied either individually or in combination in the PSA. In any case, for ensuring that probabilities are derived in a consistent manner across the wide range of events and phenomena evaluated in the Level 2 PSA, a set of rules should be developed and included in the PSA documentation. Such rules should include the rationale used to assign particular probabilistic estimates.

GROUPING OF END STATES OF ACCIDENT PROGRESSION EVENT TREES INTO RELEASE CATEGORIES

9.16. Once the end states of the individual accident progression event trees have been identified, they should be grouped into specified release categories. Since this involves the grouping of a large number (typically thousands) of end states of the accident progression event tree into a small number (typically tens) of release categories, a systematic process should be applied to this grouping process. This should be normally done using a computerized tool because of the necessity for efficiently handling a large amount of information. The particular way that this is done will depend on the software used for quantification of the accident progression event tree, but it can involve post-processing of the end states of the accident progression event tree (cutsets) or including the attributes in the accident progression event tree model and using them in the grouping process.

9.17. End states of the accident progression event tree grouped in a release category are expected to have similar radiological release characteristics and off-site consequences, so that the source term analysis carried out for the group characterizes the entire set of end states within the group and reduces the amount of source term analysis that needs to be carried out (see Section 10).

## 10. SOURCE TERM ANALYSIS FOR SEVERE ACCIDENTS

10.1. This Section provides recommendations on release categories specification and source term analysis. The extent to which source term analysis needs to be carried out depends on the objectives and intended applications of the PSA. If the source term is to be used in a Level 3 PSA, the characteristics of the environmental source term may need to be more extensive. On the other end of the spectrum, only the frequency of accidents that would result in a large early release may need to be characterized. The following recommendations can therefore be adapted according to the objectives of the PSA.

10.2. The development of source terms[23] analysis for severe accident consists of two sub-tasks, definition of release categories[24] and calculation of radiological release for each of these categories. In the first sub-task, a set of release categories to be attached to the endpoints of the accident progression event tree sequences is defined. These release categories are each identified by a set of characteristics that impact the amount of radiological release that will arise for accident progression event tree sequences matching these characteristics (see paras

[23] The term 'source term' is to be understood as defined in the IAEA Nuclear Safety and Security Glossary [46] as "The amount and isotopic composition of radioactive material released (or postulated to be released) from a facility. Used in modelling releases of radionuclides to the environment, in particular in the context of accidents at nuclear installations or releases from radioactive waste in repositories." . In addition, other definition providing more details is "The characteristics of a radionuclide release at a particular location including the physical and chemical properties of released material, release magnitude, heat content (or energy) of the carrier fluid, location relative to local obstacles that would affect transport away from the release point, and the temporal variations in these parameters (e.g., time of release, duration, etc.)", as defined in Ref. [47].
[24] A group of accident progression sequences that would generate a similar source term to the environment. The categories are defined by attributes in relation to the release.

10.5-10.6). The process of defining these release categories may be supported by code calculations and may be an iterative process. Some methodologies for Level 2 PSA involve a two-step definition of the release categories, in which an initial set of detailed release categories is analysed and then regrouped into a smaller set of release categories on the basis of similarity of radiological release (see para 10.7).[25]In the second sub-task, the calculation of radiological release for each release category, code calculations are performed for one or more representative sequences for each category. The representative sequences are defined by reference to the accident progression event tree sequences that contribute to each release category and are usually selected after quantification of the accident progression event tree so that sequences can be ranked by frequency to assist the selection process (see Section 9).

10.3. All potential plant specific release paths should be identified in the accident progression event tree and considered in the corresponding end states. For practical reasons, in accordance with FIG. 1, the end states of the accident progression event tree are generally grouped into release categories (with similar properties regarding releases). The source term analysis is then carried out only for a representative severe accident scenario of each release category. Preliminary list of representative severe accident scenario should be based on severe accident scenario established for identified PDSs (see para 6.8). The choice of representative scenarios should be justified. It is good practice to carry out sensitivity studies for the choice of representatives scenarios.

10.4. Hence, the source term analysis in Level 2 PSA involves:

(a) Defining the release categories;
(b) Grouping the end states of the accident progression event tree into the release categories;
(c) Carrying out the source term analysis for the release categories.

SPECIFYING AND GROUPING RELEASE CATEGORIES

10.5. The analyst should consider events, depending on the reactor technology, that during accident progression modelled in Level 1 or Level 2 PSA event trees have a significant influence on the release of radioactive material from the containment, for example:

(a) The mode and time of failure of the reactor core cooling;
(b) The mode and time of failure of the reactor vessel or primary circuit, and the vessel pressure at this time;
(c) The mode and time of failure of the containment (failure location, size and resulting transport pathway to the environment);
(d) The availability of systems (cooling water) and the efficiency of physical mechanisms for cooling molten core material (considering depth and composition of ex-vessel core debris);
(e) The availability of safety systems able to reduce radioactive releases (e.g. containment spray system, filtered containment venting system, suppression pool, ice condensers);

[25] In these methodologies, the term ‘source term category’ may be used to refer to the second, smaller, set of grouped release categories; in most methodologies, however, the terms ‘release category’ and ‘source term category’ are synonymous.

(f) The retention mechanisms for radioactive material (e.g. pool scrubbing, retention in pipes, filters).

10.6. After identification of all events that influence the radioactive release, a set of attributes used to characterize the release categories should be considered. Typical attributes are shown in TABLE 7 for water cooled reactors. The release of radioactive material to the environment is a function of these attributes.

TABLE 7. EXAMPLES OF TYPICAL ATTRIBUTES USED FOR THE SPECIFICATION OF RELEASE CATEGORIES FOR WATER COOLED REACTORS.

| Release attributes | Variations | |
|---|---|---|
| Time frame of the severe accident in which the containment damage/bypass first occurs | At the onset of core damage (e.g. bypass of the containment)<br>Early (during in-vessel core damage)<br>Intermediate (immediately following breach of the reactor pressure vessel)<br>Late (several hours after breach of the reactor pressure vessel) | |
| Pressure of reactor pressure vessel during core damage | High (near nominal)<br>Low (depressurized) | Typical attributes of Level 2 PSA |
| Containment pressure | High<br>Low (depressurized) | |
| Modes or mechanisms of containment leakage (associated with a time frame) | Design basis conditions leakage<br>Beyond design basis conditions leakage<br>Catastrophic rupture of containment<br>Loss of coolant accident in interfacing system<br>Steam generator tube/tubes or header rupture<br>Open containment isolation valves<br>Open material hatch access<br>Basemat penetration | |
| Active engineered features providing capture mechanisms for radioactive material | Sprays<br>Fan coolers<br>Filtered vents<br>Others (e.g. water management, reactor coolant system depressurization) | |
| Passive engineered features providing capture mechanisms for radioactive material | Secondary containments<br>Reactor buildings<br>Suppression pools<br>Overlying water pools<br>Ice beds<br>‘Tortuous’ release pathways<br>Submerged release pathway<br>Alkaline materials | |
| Phenomena | Molten Core Concrete Interaction<br>Deposition of aerosols<br>Resuspension of deposed aerosols by energetic phenomena<br>Specificities for chemical process (e.g. iodine, ruthenium,) | |

| Release attributes | Variations | |
|---|---|---|
| Time elapsing since the start of the severe accident | Short (e.g. for pressurized water reactor typically less than 2 hours)<br>Medium (e.g. for pressurized water reactor typically between 2 and 10 hours)<br>Long (e.g. for pressurized water reactor typically greater than 10 hours) | Additional attribute for linking to Level 3 PSA |
| Location of release | Ground level<br>Elevated | |
| Energy of release | Low (minimal buoyancy in ex-plant atmosphere)<br>Energetic (highly buoyant) | |
| Release rate | Rapid 'puff' release<br>Slow continuous release<br>Multiple plumes | |
| Containment failure size | Sizes proposed in square meters | |
| Source Term | Amount and composition of different radioactive nuclides / nuclide groups<br>Duration of the release (e.g. release occurring during X hours) | |

10.7. These attributes can be used to specify the set of release categories used for the source term analysis in the Level 2 PSA. For some methodologies for Level 2 PSA this process may generate a very large initial number of release categories, that should be further grouped into a manageable final set that can be used in the source term analysis through an 'integral' computer code (see paras 10.14-10.20). This grouping process can be less condensed if a mechanistic source term code is used for the source term analysis (see paras 10.21-10.25).

10.8. Some accident scenarios can include several containment failure modes. The analyst should pay attention to the quantification of the frequency of each containment failure individually in order to capture their importance on the global results.

## SOURCE TERM ANALYSIS APPROACHES

10.9. In Level 2 PSA, the source term specifies, for a given accident scenario, the amount and composition of radioactive material released from the plant to the environment and the timing, location and kinetic energy of the release. Many plant design features and accident phenomena have been recognized to affect the magnitude and characteristics of source terms for severe accidents. These include fixed plant design characteristics, such as configuration of the fuel and the control assembly and material composition, core power density and distribution, fuel burnup and concrete composition as well as radioactive decay of radioactive releases. These plant design characteristics will be the same for all the end states of the accident progression event tree. The analyst should be familiar with the specific plant design features (see Section 4) and accident phenomena (see Section 6) for the definition of end states of the accident progression event tree.

10.10. Depending on the reactor technology, for the source term calculations, a combination of the listed approaches could also be used:

(a) Applying one of the ‘integral’ computer codes such as described in Annex I for water cooled reactors to a limited number of representative accident scenarios;
(b) Applying a fast-running source term code (see ‘dedicated’ code in Annex I) to a large number of accident scenarios;
(c) Applying detailed models or bounding estimates of source terms or transposition from another NPP to obtain preliminary results such as during the design phase.

10.11. The extent to which source term analysis needs to be carried out depends on the objectives and intended applications of the PSA. If the source term is to be used within Level 3 PSA, the characterization of the source term should be sufficiently detailed to be adequate as an input for Level 3 PSA consequences calculations (e.g. see Refs [48], [46], [49]). The justification of the adequacy should be developed and documented. The analysis of off-site consequences will necessitate a detailed characterization of the release of radioactive material (i.e. a quantitative tracking of the core inventory of radioactive material at a detailed level) (see Ref. [50]).

10.12. Noble gases, iodine and caesium are often selected as leading indicators of the overall radiological source term. Thus, there are many ways of specifying the attributes of a radiological source term[26], including that different release categories may have the same source term (i.e. same amount and composition). However, it is important to specify these attributes at the beginning of the Level 2 PSA project (see 2.17-2.19).

**Source term analysis with a plant specific approach**

10.13. One option is to perform plant specific source term analysis for each of the release categories. The analysis could be conducted by using an integral computer code or a fast running/dedicated computer code (see Annex I).

*Source term analysis with an integral computer code*

10.14. One option is to use an integral computer code to perform plant specific source term analysis for each of the release categories. This code should be capable of modelling the integrated behaviour of severe accident phenomena: thermohydraulic response of the reactor, heat-up of the core, fuel damage and relocation of core material, conditions in the containment and adjacent buildings, release of radioactive material from the fuel and transport of radioactive aerosols and vapour through the reactor coolant system into the containment and subsequently to simulate the source term to the environment[27].



[26] The way the attributes are specified is also influenced by the objectives of the Level 2 PSA, for example, whether or not a Level 3 PSA or part of Level 3 PSA will be performed.
[27] Some Level 2 PSAs have developed parametric source term models on the basis of calculations performed with codes such as MAAP [51] or MELCOR [52] and this approach enables the uncertainties in the source term parameters to be combined with the integrated process for uncertainty assessment and uncertainty propagation.

10.15. In the source term analysis, all the processes that affect the release and transport of radioactive material inside the containment and in adjacent buildings should be modelled, such as:

(a) Releases of radioactive material from the fuel during the in-vessel phase;
(b) Retention of radioactive material within the reactor coolant system;
(c) Releases of radioactive material during the ex-vessel phase;
(d) Retention of radioactive material inside the containment and adjacent buildings;
(e) Resuspension, re-vaporisation, condensation and re-entrainment mechanisms (e.g. energetic phenomena, chemical reactions, mechanical effects).

10.16. Considering previous processes, in the calculation of the source term and the plant model, the spatial distribution of the radionuclide species within the reactor coolant circuit and the containment should be estimated, as well as the quantity released to the environment.

10.17. The analysis should be carried out for a representative accident sequence in each release category. Sensitivity analyses should be performed to provide confidence that the source terms have been accurately characterized and there is not an undue variation of the source term magnitude within each release category.

10.18. Work has been carried out on calculating the releases within a dynamic PSA framework (see Ref. [49]) where the releases and the accident progression area calculated together in an integral manner.

10.19. Source term calculations with integral computer codes for severe accident analysis, generally consider group categories of radioactive elements or chemical compounds rather than on individual radioisotopes [53], [54]. This simplification is necessary to reduce the hundreds of radioisotopes of radioactive material and actinides generated in nuclear reactor fuel to a reasonable number of groups of radioactive elements that can be tracked by an integrated severe accident computer code. Different group categories have been used in different computer codes. However, most group categories are based on similarities in the physical and chemical properties of the radioactive elements. The group category also takes into account similarities in the chemical affinity of the elements to reactions with other radioactive elements and non-radioactive material that they might encounter in transport within the reactor coolant circuit and containment, e.g. steam, hydrogen, structural materials. A typical group category used in the analysis of releases of radioactive material is shown in TABLE 8. The source term, therefore, could be expressed in terms of the fraction of the initial core inventory of one or more of these groups of radionuclides. Concerning refuelling outage related operating modes, the stage of refuelling (before/after) and the subsequent mixture of newer and older fuel elements should be considered in the definition of the core inventory. The analyst should be familiar with the composition of radioactive material in the group categories proposed by the integral computer code used.

10.20. The efficiency with which the groups of radionuclides listed in TABLE 8 are transported to the environment depends strongly on the chemical form they assume after leaving the core region. Numerous chemical interactions can occur, which cause elemental forms of these species to react and form compounds with a wide range of physical properties [55]. Iodine, for example, is widely known to react with caesium to form volatile caesium iodide. However, this is not the only form in which iodine can be transported along the release pathway. Several species listed in TABLE 8 can be transported in more than one chemical form. Partitioning of the core inventory of reactive species among their possible chemical forms is considered as a best practice that may introduce a source of uncertainty in the source term calculations. If performed, the impact of the partitioning in the assessment of source term calculations should be considered, for example, via a sensitivity study.

TABLE 8. EXAMPLES OF GROUP CATEGORIES FOR ELEMENTS IN RADIOACTIVE MATERIAL

| Group | Elements in group | Representative element in group |
|---|---|---|
| Noble gases | Xe, Ne, Ar, Kr, He, Rn, H, N | Xe |
| Halogens (aerosols) | I, Br, Cl, F | I |
| Halogens (gaseous) | $I_2$, $Br_2$, $Cl_2$, $F_2$ | $I_2$ |
| Halogens (organic) | $ICH_3$, $BrCH_3$ | $ICH_3$ |
| Halogens (oxidized) | $IO_x$, $BrO_x$ | $IO_x$ |
| Alkali metals | Cs, Rb, Li, Na, K, Fr, Cu | Cs |
| Alkaline earths | Ba, Mg, Ca, Sr, Be, Ra, Es, Fm | Ba |
| Chalcogens | Te, S, Se, Po | Te |
| Refractory metals | Ru, Mo[a], Pd, Tc, Rh, Re, Os, Ir, Pt, Au, Ni | Ru |
| Lanthanides | La, Y, Nd, Eu, Pm, Pr, Sm | La |
| Actinides | Ce, Pu, Np, Zr, U[a], Ti, Zr, Hf, Th, Pa, C | Ce |
| Trivalents | La, Al, Sc, Y, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf | La |
| More volatile main group | Cd, Hg, Zn, As, Sb, Pb, Tl, Bi | Cd |
| Less volatile main group | Sn, Ga, Ge, In, Ag | Sn |
| Boron | B, Si, P | B |

[a] Mo and U are represented as separate groups in some models.

*Source term analysis with a fast-running code*

10.21. A second option is to use a fast-running computer code to perform plant specific source term analysis with no limitation on the number of calculations. Such a code does not calculate the integrated behaviour of severe accident phenomena (e.g. thermohydraulic response of the reactor, core melt) but calculates only the release of radioactive material from the fuel and transport of radioactive aerosols and vapour through the reactor coolant system into the containment, the behaviour of radioactive materials in the containment and the release outside. The information on thermohydraulic response, fuel melt, conditions in containment, energetic phenomena or accident kinetics are available through the release categories attributes and are input data for the fast-running code.

10.22. For fast-running codes the grouping of radionuclides is defined in a similar way as in integral code (see TABLE 8).

10.23. A fast-running code should be validated for example by comparison with an integral code or with experimental data.

10.24. Uncertainties on key parameters related to the accident phenomena mentioned in the last sentence of para. 10.21 should be considered in the fast-running code to allow for the quantification of uncertainties in the source term analysis.

10.25. A fast-running code could be integrated in the accident progression event tree so that the accident progression event tree quantification includes the source term analysis including uncertainties.

**Source term analysis with a simplified approach**

10.26. Another option is to use the source term analysis from another plant where the design and features of the reference plant relating to the progression of severe accidents are sufficiently similar to the plant being analysed and the results of the deterministic analysis are available. When reference studies are used as a surrogate for plant specific calculations, it is important to note that three criteria should be met for reference plant analyses to be acceptable for use in a Level 2 PSA:

(a) A technical basis should be established to justify that the plant under study is sufficiently similar to the proposed reference plant. Design features that affect the transport of radioactive material and its retention within the reactor pressure vessel, associated coolant system piping and containment structures/systems should be identified and compared.
(b) It should be ensured that the accident sequence(s) modelled in reference plant analysis are sufficiently similar to the accident sequences of interest to the Level 2 PSA for the plant under investigation. Differences in the operation of reactor safety systems or containment systems can invalidate the applicability of a reference plant calculation to a particular PDS[28].
(c) The reference plant calculation should be performed using a contemporary model of plant response to severe accident phenomena. Caution should be used in applying reference plant results that are several years old. The state of knowledge and level of sophistication in modelling the progression of severe accidents have evolved significantly in recent years and thus reduced the value of some results available in the open literature (i.e., scientific and technical publications).

10.27. Use of the source term analysis from another plant may be helpful to define parametric models or bounding estimates of source terms, especially during the design phase of a new nuclear power plant.

[28] For example, many calculations of accident sequences involving ‘station blackout’ for several reactor designs can be found in the open literature. However, there are many variations of station blackout, depending on the particular system configuration of a plant. In some cases, sufficient direct current power might be available to operate a small group of components (e.g. relief valves) or systems (e.g. steam driven pumps) in some plants that are not available in other plants. Such differences should be carefully considered before calculated results from the literature are applied to the plant under study.

USE OF COMPUTER CODES FOR SOURCE TERM ANALYSIS

10.28. In line with Requirement 18 of GSR Part 4 (Rev.1) [2], particular models and correlations introduced in the computer codes used for the source term analysis (i.e. integral or fast-running codes) should be verified and validated.

10.29. The users of the computer code for source term analysis should be trained in the use of the code and be familiar with the phenomena being modelled by the code and the way that they interact, the meaning of the input and output data, and the limitations of the code. Other recommendations are provided in paras 3.15 to 3.17.

RESULTS OF THE SOURCE TERM ANALYSIS

10.30. The overall results of the source term analysis should be clearly presented and documented. The characteristics of the source terms associated with the release categories should be clearly documented. One way of doing this is to present the results in the form of a matrix similar to the C matrix described in Section 11, in which the frequency (or the contribution to the total core damage frequency) of each release category is tabulated. An example format for presenting the results of the source term analysis is shown in TABLE 10 (another example is presented in Ref. [56]).

10.31. The source terms and frequencies of the release categories, the later obtained as a result of accident progression event tree quantification, should be used to determine the large release frequency or the large early release frequency for comparison with numerical probabilistic safety goals or criteria where they have been set, as described in Section 11. This will require the terms 'large' and 'early' to have been defined within the Level 2 PSA project. This can be done in a number of ways, as outlined in paras 2.16-2.19 and in Ref. [56]).

10.32. An alternative format for displaying the results of the source term analysis is by means of a complementary cumulative distribution function that is based on the frequency of releases greater than X, where X varies from the smallest to the largest calculated quantity of release. This will require the term 'quantity of release' to be defined within the Level 2 PSA project, which might be understood, for example, as the activity of a leading radioisotope or of a group of relevant radioisotopes. The frequency of releases and the magnitude of releases should be considered together for the interpretation of the Level 2 PSA and its applications.

10.33. The insights gained from such a quantitative evaluation of radionuclide releases should be summarized and discussed. The results of the quantitative sensitivity analysis or uncertainty analysis should also be presented and discussed. In particular, for each radioactive material group, the frequency of exceeding a given release quantity should be provided. The results should clearly show the statistical significance of each complementary cumulative distribution function (e.g. mean, median, 95th percentile).

ANALYSIS OF UNCERTAINTIES

10.34. Requirement 17 of GSR Part 4 (Rev.1) [2] states that "**Uncertainty and sensitivity analysis shall be performed and taken into account in the results of the safety analysis and the conclusions drawn from it.**" Uncertainty and sensitivity analyses help to understand how the various modelling options within a code affect calculated results. In addition to the uncertainties in modelling severe accident phenomena, many of the chemical and physical processes governing the release of radioactive material from fuel, deposition and retention on reactor internal surfaces, containment surfaces, and from scrubbing by containment safety systems are still subject to research. Major sources of uncertainty in the evaluation of source terms are listed in TABLE 9.

10.35. Past and ongoing research programmes have made significant progress towards reducing uncertainty in severe accident source terms (e.g. Refs [53], [54]). Uncertainties associated with the physical processes involved in core damage and core relocation lead to uncertainty in respect of the release of radioactive material from fuel (see Section 6). Uncertainties associated with containment response to beyond design basis conditions lead to uncertainty in respect of the driving forces for radioactive material transport along the pathway to the environment. Examples of uncertainties associated with these areas are given in Section 7.

10.36. The Level 2 PSA should represent the up-to-date knowledge on severe accidents and on fission products behaviour. The assessment of uncertainties can be addressed by carrying out sensitivity studies for the major sources of uncertainty that influence the results of the Level 2 PSA (see paras 11.26-11.27). Uncertainties modelling can be also introduced directly in the accident progression event tree (distribution of probability) for their propagation inside the model, while it is possible depending on the PSA tool.

TABLE 9. EXAMPLES OF ISSUES GIVING RISE TO UNCERTAINTIES IN SOURCE TERMS FOR WATER COOLED REACTORS

- Uncertainties in core damage processes and containment behaviour (see Sections 6 and 7);
- Effects of fuel exposure (burnup) on the release fraction of radioactive material from fuel matrix;
- Chemical forms of volatile and semi-volatile species;
- Chemical interactions with fuel, neutron absorbers and structural materials during core degradation;
- Deposition rates of radioactive material and aerosols on the surfaces of the reactor coolant circuit;
- Deposition of radioactive material in piping and other components in accident sequences with containment bypass;
- Release of radioactive material and aerosols during molten core–concrete interaction
- Chemical processes during molten core–concrete interaction;
- Interaction between hydrogen burn or radicals in flame fronts and airborne radioactive material (e.g. possible resuspension of radioactive deposits);
- Scrubbing efficiency of aerosols and vapours in suppression pools, ice beds or bubble towers;
- Aqueous chemistry of radioactive material captured in water pools;
- Re-vaporization and resuspension of radioactive material from surfaces;
- Chemical decomposition of radioactive material aerosols;
- Radioactive release into the environment with regard to containment break size, containment leak rate, released fraction of inventory, iodine chemistry.

*Note:* Many topics above concern iodine (and also ruthenium) forms and behaviour. Due to the importance for radiological consequences assessment, uncertainties reduction is an issue for dedicated severe accident research programmes.

TABLE 10. EXAMPLE SUMMARY OF SOURCE TERMS FOR WATER COOLED REACTORS

| | Release categories attributes | | | | | | | Fraction of core inventory to environment[a] | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | Release category | Frequency ($a^{-1}$) | Time release begins | RCS[b] pressure at vessel failure | Mode of containment leakage | Release through auxiliary building | Active attenuation mechanism | Xe | I | Cs | other |
| Source terms | 1 | i.iE-i | Early | Low | SGTR[c] | Yes | None | 0.995 | 0.11 | 0.08 | x.xE-x |
| | 2 | j.jE-j | Intermediate | High | Rupture | No | None | 0.99 | 0.14 | 0.11 | y.yE-y |
| | : | | | | | | | | | | |
| | : | | | | | | | | | | |
| | X | k.kE-k | Intermediate | Low | Nominal leakage | Yes | Sprays | 0.84 | 0.04 | 0.02 | i.iE-i |
| | : | | | | | | | | | | |
| | : | | | | | | | | | | |
| | Y | m.mE-m | Late | Low | Rupture | No | Sprays | 0.89 | 0.002 | 0.001 | j.jE-j |
| | : | | | | | | | | | | |
| | N | | | | | | | | | | |

[a] These are sample values only.
[b] RCS: reactor coolant system.
[c] SGTR: steam generator tube rupture.

## 11. QUANTIFICATION OF EVENT TREES AND ANALYSIS OF RESULTS

### QUANTIFICATION OF EVENT TREES

11.1. The quantification process consists of calculating the frequencies of the end states of the accident progression event tree. The results of this quantification lead to basic results of Level 2 PSA and the basic results can be presented by different groupings of end states (e.g. containment failure modes, type of releases, kinetics, source terms for Level 3 PSA).

11.2. The quantification process depends on the PSA computer codes used for the development of the accident progression event trees. Two categories of PSA computer codes exist based on the minimal cutsets calculations (Boolean algorithm) or on scenario calculation (chronological algorithm). For both categories, the frequency of the release categories or other groupings are calculated by aggregating the frequencies of all the end states of the accident progression event tree that are assigned to the group, depending on the PSA computer code used.

11.3. The Level 2 PSA quantification process may be accomplished in various formats with either direct or intermediate links between the initiating event sequences and the ultimate Level 2 PSA release categories. For the approach using a combination of small event trees and a large fault tree (the fault tree linking approach, see Ref. [34]), Boolean reduction is carried out by the software for the logic models developed using event trees and fault trees for each initiating event group. As with the Level 1 analysis, before quantifying the Level 2 PSA, care should be taken to ensure that no logic loops exist in the model. If such loops exist, breaking the loops is a prerequisite for quantification.

11.4. The probabilistic quantification of the Level 2 PSA should be carried out using a suitable computer code that has been fully validated and verified. Most PSA computer codes used for the Level 1 PSA are suitable for Level 2 PSA analysis as well. In performing these analyses the users of the codes should be adequately experienced in PSA modelling and should have a detailed understanding of the severe accident progression process and understand the limitations of the code.

11.5. The overall results of the quantification of the Level 2 PSA model should include:

(a) Release categories frequency;
(b) Contributions to the release categories frequency arising from each of the Level 1 PSA PDSs (or if directly linked to the Level 1 PSA, the initiating event groups);
(c) Cutsets and cutset frequencies (for the fault tree linking approach) or scenarios and scenario frequencies (for the approach using event trees with boundary conditions);
(d) Contribution of significant accident progression event tree sequences to the release frequencies;
(e) Results of sensitivity studies and uncertainty analysis;
(f) Importance measures (such as the risk achievement worth and the risk reduction worth for basic events) that are used for the interpretation of the Level 2 PSA.

11.6. For the purpose of general verification of the correctness of the severe accident progression sequences modelling, the validation of release categories frequencies sum against the core damage frequency determined from Level 1 PSA (typically core and/or fuel damage frequency) should be done. Justification for any numerical deviations should be given.

11.7. The analysts should check that the accident sequences or cutsets identified by the solution of the Level 1 PSA model are propagated into the Level 2 structure and are appropriately reflected in the release categories. In addition, a check should be made to confirm that the cutsets (or sequences) representing combinations of initiating events, component failures, and severe accident phenomena that are expected to lead to containment failure are indeed included in the list of cutsets (or sequences) generated. The software used for accident progression event tree quantification should be capable to correctly quantify success branches in the event tree. The reason is that success branches in accident progression event tree are equally important as failed branches. Actually, the meaning of success and failure in accident progression event tree does not always apply – both are alternatives of the process and both may have high probability.

11.8. Taking into consideration the definition of all risk metric terms used to characterize the significance of containment failure and releases that should be defined at the beginning of the Level 2 PSA study (see 2.16-2.19), containment failure and source term associated release metrics could take the form of an absolute criterion or a relative criterion (e.g. relative to total core damage frequency or large release frequency, see Annex III).

11.9. A check should be made that any post-processing that has been carried out on the Level 2 PSA cutsets to remove mutually exclusive events or to introduce recovery actions not included explicitly in the Level 2 PSA model, has produced the correct results. Depending on the PSA code used and the quantification options applied, it is possible that post-processing of cut-sets may introduce inconsistencies in the numerical release frequency results leading to discrepancies between the total Level 1 and Level 2 PSA results. The PSA analyst should be aware of these possibilities and should check whether the Level 2 PSA conserves the frequency input from Level 1 PSA.

11.10. For quantification of the Level 2 PSA, truncation limits (e.g. cut-offs) will need to be specified to limit the time taken for the analysis. The usual approach is to set a truncation frequency limit so that cutsets with a lower frequency that would represent a negligible contribution to risk may be confidently omitted from the final quantification. Justification should be provided that the truncation limit has been set at a sufficiently low level that the overall result from the Level 2 PSA converges and the chosen limit has a negligible impact on the estimated frequency of a large release. The choice of cut-off may vary depending on the application of the Level 2 PSA. Performance of a convergence study is a typical way to fulfil this requirement. In a convergence study the analyst will perform quantification at a number of cut-off values to identify a cut-off value at which a stable frequency result is obtained.

## ANALYSIS OF RESULTS OF ACCIDENT PROGRESSION EVENT TREES

11.11. Results and insights gained from the quantification of accident progression event trees should be summarized and discussed. In the first place, the frequencies of each release categories should be indicated.

11.12. In addition, useful information and insights gained through Level 2 PSA quantification should also be indicated. For this purpose, results can be presented in various formats. They are often tabulated in the form of a so-called containment performance matrix ('C matrix'), which is a concise way of comparing the relative likelihood of the various outcomes of the accident progression event trees. The C matrix identifies the PDS frequency, F(m), and the conditional probabilities C (m, n) that a release category 'n' can be realized, given a PDS 'm' (see Equation 1). An example layout for the content of a C-matrix is presented in TABLE 11. Uncertainty analysis leads to alternative sets of values of the elements of the C matrix. The C matrix is structured such that the release frequency, R(n), for each release category, n is given by:

$$R(n) = \sum_{m=1}^{M} F(m)\, C(m,n)$$

Equation 1. C matrix for each PDS frequency, F(m), and the conditional probabilities C (m, n) that a release category 'n' can be realized, given a PDS 'm'.

TABLE 11. MITIGATION PERFORMANCE MATRIX ('C MATRIX')

| PDS | Release category | | | | | | | PDS frequency |
|---|---|---|---|---|---|---|---|---|
| | 1 | 2 | .. | n | .. | .. | N | |
| 1 | C(1,1) | C(1,2) | .. | C(1,n) | .. | .. | C(1,N) | F(1) |
| 2 | C(2,1) | C(2,2) | .. | C(2,n) | .. | .. | C(2,N) | F(2) |
| 3 | C(3,1) | C(3,2) | .. | C(3,n) | .. | .. | C(3,N) | F(3) |
| .. | .. | .. | .. | .. | .. | .. | .. | .. |
| m | C(m,1) | C(m,2) | .. | C(m,n) | .. | .. | C(m,N) | F(m) |
| .. | .. | .. | .. | .. | .. | .. | .. | .. |
| M | C(M,1) | C(M,2) | .. | C(M,n) | .. | .. | C(M,N) | F(M) |
| Release category frequency | R(1) | R(2) | .. | R(3) | .. | .. | R(N) | R |

11.13. For each PDS, the main minimal cutsets could also be indicated. A variant to the C matrix would be to identify the conditional probabilities C (m, n) that a release category 'n' can be realized, given the families of PDS, B (j, m) for the list of initiating events A(j) (see Equation 2). This may not be sufficient and additional useful information may be presented such as the release categories frequencies for each plant operating state, the distribution of the different causes of containment failure for specific release categories.

$$R(n) = \sum_{m=1}^{M} \sum_{J=1}^{J} A(j)\, B(j,m)\, C(m,n)$$

Equation 2. C matrix for the conditional probabilities C (m, n) that a release category ‘n’ can be realized, given the families of PDS, B (j, m) for the list of initiating events A(j).

11.14. The major contributors to each release category of interest should be identified and explained. This generally concerns large releases. However, this approach can also be extended to any other consequence deemed necessary. The root causes of variations in the conditional probability of each examined consequence among the various PDSs should be explored and explained.

11.15. As discussed above, by combining the results of the Level 1 PSA (frequencies of occurrence of the various PDSs and their associated uncertainties) with the conditional probabilities of various containment failure modes and/or release modes and their associated uncertainties resulting from quantification of the accident progression event tree; the frequencies and uncertainties associated with each release category can be determined. It is useful to summarize, the contribution of each release category (R(n)) to the large release (early large release) frequency, R, should also be tabulated, to enable identification of major contributors to the total release frequency.

11.16. For each of the selected release categories, or related group of release categories, one representative accident sequence is selected for which a source term is estimated on the basis of results obtained from plant specific calculations employing an appropriate computer code for estimating source terms for severe accidents, as discussed in Section 0 and Annex I, or past analyses from Level 2 PSAs of representative plants. When using representative plant analyses for releases care should be taken to account for plant differences in core fission product inventory (typically associated with fuel design, core power and operational history, and containment failure modes and failure pressures). Considerations regarding the acceptability of source terms from representative plant specific PSAs should be documented.

11.17. The selection of the representative accident sequence should be governed by its frequency and consequence dominance within the release category. Alternatively, source terms can be estimated for each and every accident sequence contributing to a particular release category. An intermediate approach is sometimes taken where calculations are performed for the dominant accident sequence and an alternative accident sequence in each release category. In addition, for release categories that result from potentially uncertain mechanisms (e.g. steam explosion, direct containment heating) for which trustworthy models might not be readily available for the code used, code calculations could be augmented by simple analyses and expert judgement.

IMPORTANCE, UNCERTAINTY AND SENSITIVITY ANALYSES

**Importance analysis**

11.18. Importance measures for basic events, groups of basic events, safety systems, groups of initiating events, etc., should be calculated and used to interpret the results of the Level 2 PSA. Importance metrics are typically focused on contributions to containment failure frequency, large early release frequency and large release frequency, however other potential end states may be of interest. These metrics may be more specific or may encompass more than one operating mode or operating state. Importance measures typically include: (a) the Fussell-Vesely importance (F-V); (b) the risk reduction worth; (c) the risk achievement worth and (d) the Birnbaum Importance metric. The various importance measures provide a perspective on which basic events, etc., contribute most to the current estimate of risk (Fussell-Vesely importance, risk reduction worth), which contribute most to maintaining the level of safety (risk achievement worth) and for which basic events the results are most sensitive (Birnbaum importance).

11.19. Importance analysis should identify contributions and impact on the risk of reactor operating modes, main SSC failures, actions from operating personnel, internal and external hazards, and mitigation strategies [10].

11.20. The importance values, as defined in 11.18, should be used to identify areas of the design or operation of the plant where improvements need to be considered.

**Types of uncertainties**

11.21. Paragraphs 11.22 through 11.26 provide recommendations on meeting Requirement 17 of GSR Part 4 (Rev.1) [2] on uncertainty and sensitivity analysis for Level 2 PSA (issues giving rise to uncertainties are presented in TABLE 4 and TABLE 9).

11.22. Uncertainty arises in a Level 2 PSA analysis as a result of several factors. The analyst should consider the following sources of uncertainties or develop, use and justify an alternative, as appropriate:

(a) *Incompleteness uncertainty.* The overall aim of a Level 2 PSA is to assess the possible scenarios (sequences of events) that can lead to releases of radionuclides, mainly those scenarios modelled in the Level 1 PSA. However, there is no guarantee that this process can ever be complete and that all possible scenarios have been identified and properly assessed. This potential lack of completeness introduces an uncertainty in the results and conclusions of the analysis that is difficult to assess or quantify. It is not possible to address this type of uncertainty explicitly. However, extensive peer review can reduce this type of uncertainty, for example by verifying the adequacy of the sequence consisted by cutsets, correctness of the input parameters, and assumption of human errors, so the Level 2 PSA should have extensive peer review. Sensitivity analyses, including bounding analyses, may be employed to provide estimates regarding the significance of the uncertainty, so the Level 2 PSA should ensure that those sensitivity analyses are performed.

(b) *Loss of detail due to aggregation.* Loss of detail can occur at several points in the Level 2 PSA analysis. Grouping accident sequences or cutsets from the Level 1 PSA into PDSs for input into the Level 2 PSA for practical reasons also introduces uncertainties due to the resulting loss of some modelling detail. Further, the process of 'binning' (or grouping) accident sequences based on key sequence attributes introduces bias and uncertainty through the possibility that the attributes used by the analyst to group 'similar' accident sequences are necessarily broad. In Level 2 analyses these grouped sequences can vary in the timing of the initial fuel release, impact of the event progression on the containment, and the magnitude and content of the containment fission product release, among others. The grouping process often has two competing goals, optimizing the number of detailed severe accident analyses to be performed while ensuring the grouping process does not distort Level 2 PSA insights or the characterization of the Level 3 PSA inputs. The impact of the binning process on uncertainty is difficult or impossible to quantify precisely; however, it is the intent of this binning process to maintain the key features of the release class such that more refined analyses or subdivision will not impact Level 2 PSA insights. In practice, the binning process is intended to be performed conservatively. It is expected as more detailed resolution regarding severe accident phenomenology becomes available and increases in computing resources allow increasing levels of detail to be captured in the PSA, this uncertainty will diminish. Sensitivity analyses may be used to assess the extent of these modelling uncertainties on the plant Level 2 response. To mitigate these uncertainties the binning within the Level 2 PSA should be carefully explained and justified and well documented and sensitivity studies should be carried out as described here before in this paragraph.

(c) *Modelling uncertainty.* This arises due to a lack of complete knowledge concerning the severe accident phenomenology, limitations related to the reproducibility of real severe accident conditions by research experiments, and appropriateness of the methods, models, assumptions and approximations used in assessment of those processes and the individual analysis tasks that support a Level 2 PSA. Modelling uncertainties are formally addressed as part of the uncertainty treatment in the Level 2 PSA with consideration of severe accident progression analysis (see paras 6.24–6.27), containment integrity analysis (see paras 7.23-7.30), human and equipment reliability assessment (see paras 8.18-8.22), development of accident progression event trees (see paras 9.7-9.12) and source term analysis (see paras 10.34-10.36).

(d) *Parameter uncertainty.* This arises due to the uncertainties associated with the values of the fundamental parameters used in the quantification of the Level 2 PSA, such as equipment failure rates and frequencies of initiating event sequences. Parameter uncertainty may also arise from uncertainty in the physical attributes of the parameters used to model the containment challenge and containment response. This is the type of uncertainty that is usually addressed by an uncertainty analysis through specifying uncertainty distributions for all the parameters and performing sensitivity studies or propagating them through the analysis. The steps for addressing the parameter uncertainties within Level 2 PSA are provided in 11.25-11.26.

11.23. Since Level 2 PSA analysts use probabilities in the accident progression event trees to reflect confidence that particular choices of modelling parameters or event outcomes are the correct ones, the Level 2 PSA is in some sense directly concerned with the treatment of uncertainties, which is therefore one of the most important aspects of the analysis.

11.24. The Level 2 PSA analysts should identify the dominant sources of uncertainty in the analysis and should quantitatively characterize the effects of these uncertainties on the baseline (point estimate) results. Characterization is typically supported using two methods: (i) sensitivity analysis and (ii) uncertainty analysis.

**Uncertainty analysis**

11.25. Whereas sensitivity analysis is used to measure the extent to which results would change if alternative models, hypotheses or values of input parameters were selected (and thus provides an evaluation of uncertainty in respect of a particular issue or a particular group of related issues at a time), uncertainty analysis examines a range of alternative models or parameter values, assigns each model or value a probability distribution and generates a distribution of the results, within which the baseline results represent one possible outcome. Each result within the full distribution is accompanied by a (subjective) probability representing the degree of belief in that result. Cumulative probability levels for the results can be calculated (e.g. the $5^{th}$, $50^{th}$ and $95^{th}$ percentiles represent 5%, 50% and 95% probabilities, respectively, and the 'true' result is below the respective level for which each of these probabilities is stated). In general, the process of quantification and propagation of uncertainties in the Level 2 PSA can be divided into four principal steps, which the analyst should carry out as follows:

(1) *Specification of the scope of the uncertainty analysis.* The sources of uncertainty in a Level 2 PSA are numerous, and it is impractical to address all of them quantitatively. Experience in performing uncertainty studies for limited aspects of severe accident phenomena suggests that the effects of uncertainties from some sources are larger and more dominant than the effects of uncertainties from other sources. In an integral sense, then, the aggregate uncertainty in Level 2 PSA results can be estimated by selecting the dominant sources of uncertainty and treating them in detail. Reference [31] provides information on an evaluation of uncertainties in relation to severe accidents and Level 2 PSA.

(2) *Characterization and/or evaluation of uncertainty issues.* After the definition of the scope of the analysis, the second step is to identify the range of values of uncertain parameters. Each value within the range of values that the uncertain parameter can take on is associated with a probability, thereby creating a probability density function or probability distribution. In many cases, such density functions or probability distributions will have been determined in the assessment of probabilities for branch points in the accident progression event tree. Judgements reflected in the probability distributions for each parameter should be supported by data, analyses and consideration of the published literature and considered for peer review as part of the independent verification (see paras 3.23-3.28).

(3) *Propagation of uncertainties*. Propagation of uncertainties are usually carried out by simulation methods based on either simple (Monte Carlo) random sampling or stratified (Latin hypercube) sampling procedures. Additional details can be found in Refs [24], [22], [57]-[64].

(4) *Display and interpretation of results.* The results of the uncertainty analysis should be carefully evaluated to strengthen the conclusions of the Level 2 PSA. In modern PSAs that include a quantitative assessment and propagation of uncertainties, the results are displayed using histograms, probability density functions, cumulative distribution functions and tabular formats showing the various quantiles of the calculated uncertainties, together with the estimates of the mean and median of the probability distributions [24], [22]. Regression analysis techniques can also be applied to assess the importance of particular uncertain issues in the PSA (e.g. Ref. [31]). Correlation coefficients of dependent variables with respect to uncertain issues or phenomena can

provide insights into their importance.

**Sensitivity analysis**

11.26. Parameter/event/phenomenon specific sensitivity analysis may be used to supplement a more comprehensive uncertainty analysis. Sensitivity analysis is a useful tool to guide the selection of dominant sources of uncertainty. Example areas of uncertainty related to the progression of severe accidents are listed in TABLE 4.

11.27. If a sensitivity analysis is used as a surrogate for a comprehensive uncertainty analysis, metrics should be developed to indicate the influence of alternative models or parameter values on the results of the Level 2 PSA.

## 12. DOCUMENTATION OF THE ANALYSIS: PRESENTATION AND INTERPRETATION OF RESULTS

### OBJECTIVES AND CONTENT OF DOCUMENTATION

12.1. Requirement 20 of GSR Part 4 [2]states that "**The results and findings of the safety assessment shall be documented.**" The primary objectives of the PSA documentation should be to meet the needs of its users and be suitable for the specific applications of the PSA. Possible users of the results of Level 2 PSA include:

(a) Operating organizations of nuclear power plants (management and operating personnel);
(b) Designers and vendors;
(c) Reviewers;
(d) Regulatory bodies and persons or organizations providing them with technical support;
(e) Other government bodies;
(f) The public.

12.2. PSA documentation includes work files, computer model inputs and outputs, correspondence, interim reports, internal reports and the reference report of the PSA, which might be or not in addition to the Safety Analysis Report. The documentation of PSA in general should be complete, well structured, clear and easy to follow, i.e. the order of appearance of analysis in the final documentation should follow, as far as possible, the order in which it was actually performed.

12.3. The documentation of the Level 2 PSA study should provide within the reference report (or by reference to available material) all necessary information to reconstruct the results of the study. The results of internal reviews, audits and peer reviews related to the Level 2 PSA study should be documented, and made available for consultation, either as part of the Level 2 PSA reference report or as part of internal reports.

12.4. The documentation of the Level 2 PSA report should provide sufficient information to satisfy the objectives of the study and to support the needs of the users of the Level 2 PSA.

12.5. To support maintaining a living PSA, in line with Requirements 12 and 24 of GSR Part 4 (Rev.1) [2], the documentation of the Level 2 PSA should also facilitate its subsequent refinement, updating and maintenance in the light of changes to plant configuration, technical advances in severe accident analysis, integration of new topics, use of improved models, broadening of the scope of the PSA in question and its use for alternative applications.

12.6. The documentation of the Level 2 PSA should provide explicit presentation of the assumptions, exclusions, limitations and features considered in the Level 2 PSA study for extending and interpreting the Level 2 PSA results, as this information is also of critical importance to users.

12.7. A fully auditable trail of calculations including, intermediate analyses, rationales for probabilistic estimates, assumptions and supporting calculations should be provided in the documentation, either as appendices or as internal reports. This is very important for reconstructing and updating each detail of the analysis in the future or for facilitating the independent review of Level 2 PSA.

12.8. Conclusions should be distinct and should reflect not only the main general results but should emphasize the conclusions drawn from the analysis of uncertainties associated with phenomena, models and databases and the supporting analyses. The effect of underlying assumptions, uncertainties and conservatisms in the analyses and methods on the results of the Level 2 PSA should be demonstrated through the presentation of the results of sensitivity studies.

12.9. If screening criteria have been applied to eliminate accident sequences with low frequencies of occurrence from further analysis, for example, from the output of the Level 1 PSA or in the definition of PDSs, then an estimate of the contribution of the truncations should be assessed and should be presented with the final Level 2 PSA results.

12.10. The Level 2 PSA report should clearly document important findings of the Level 2 PSA, including:

(a) Plant specific design or operational vulnerabilities identified;
(b) Key operator actions for mitigating severe accidents;
(c) Potential benefits of various engineered safety features;
(d) Areas for possible improvement in operations or hardware for the plant and the containment in particular.

12.11. The results of the PSA may be compared with probabilistic safety criteria for Level 2 PSA, if these have been set. Available probabilistic safety criteria and/or goals vary considerably among Member States, but the most common risk metrics for Level 2 PSA include criteria and/or goals for the frequency of a large early release and the maximum tolerable frequency of releases of various magnitudes (see paras 2.16 to 2.19 and Annex III). While the threshold for large early release frequency represents a point estimate frequency for a particular unacceptable release, the maximum tolerable frequency of releases of various magnitudes expands this concept across the full range of possible releases.

12.12. Some parts of the documentation may be intended for use within the operating organization, while other parts of the documentation may be intended for wider external use. Some of the users, for example the public, might use, primarily, the summary report of the PSA, while others might use the full PSA documentation, including the computer model. The nature and the amount of information for inclusion in the documentation for external use compared with that intended for in-house support documentation should be established in accordance with the policies and process defined in the operating organization, this decision-making process may include the PSA team and the project management for the Level 2 PSA.

ORGANIZATION OF THE DOCUMENTATION

12.13. Recommendations on the organization and preparation of documentation for PSA are provided in SSG-3 (Rev. 1) [4] and are equally applicable to Level 2 PSA. This Section provides specific recommendations on documenting the results and findings of Level 2 PSA. The Level 2 PSA documentation should be divided into three major parts, namely:

(1) Summary report;
(2) Main report;
(3) Appendices to the main report.

12.14. The summary report should be designed to provide an overview of motivations, objectives, scope, assumptions, results and conclusions of the Level 2 PSA and potential impacts on plant design, operation and maintenance. The summary report generally is aimed at a wide audience of reactor safety specialists and should be adequate for high level review. Other aspects of the summary report are described in SSG-3 (Rev. 1) [4].

12.15. An outline of the main report should also be provided in the summary report, to guide reviewers to sections where additional details and supporting analyses related to severe accident progression, human performance, equipment reliability and containment integrity analyses are included. The summary report should be prepared by an individual who has an excellent overview of the entire PSA study. It should be independently reviewed by individual task leaders and/or analysts for correctness and consistency.

12.16. The summary report of a Level 2 PSA should include a subsection on the structure of the report, which should present concise descriptions of the contents of the sections of the main report and of the individual appendices. The relation between various parts of the PSA should also be included in this subsection of the summary report.

12.17. The main report should give a clear and traceable presentation of the complete Level 2 PSA study, including:

(a) A description of the plant at the moment of the study;
(b) The objectives, scope and approach of the study;
(c) The methods and data used;
(d) The grouping of accident sequences from Level 1 PSA, the PDSs considered as well as the screening criteria for the final set of PDSs;
(e) The assumptions and results related to severe accident progression regarding the modelling of phenomena, the containment strength analysis, the human and equipment reliability modelling and the accident progression event tree;
(f) The results and the conclusions of the Level 2 PSA study documenting the source term, the risk metrics and the uncertainties, sensitivity and importance analyses.

12.18. The main report, together with its appendices, should be designed:

(a) To support technical review of the Level 2 PSA;
(b) To communicate key detailed information to interested users;
(c) To permit the efficient and varied application of the Level 2 PSA models and results;

(d) To facilitate the updating of the models, data and results in order to support the continued safety management of the plant.

12.19. The appendices should contain detailed data, records of engineering computations and detailed models. The appendices should be structured so as to correspond directly to the sections and subsections of the main report, as far as possible.

12.20. Details of the rationale and analyses employed for a Level 2 PSA should be reported in a way that presents information on the methods used, the PSA process, and the insights and conclusions drawn in a logical manner. The report should be compiled in such a way that it facilitates review activities, including peer review, and provides a structured entry route to detailed supporting material.

12.21. A sample outline for the documentation for a Level 2 PSA is provided in Annex II.

COMMUNICATION OF RESULTS

12.22. Paragraph 12.22 to 12.23 present the recommendations to meet the Requirement 24 of the GSR Part 4 (Rev. 1) [2] related to the maintenance of the safety assessment. The communication of Level 2 PSA results should be in accordance with the possible radiation risks from the facility and considering the complexity for modelling the severe accident phenomena and the limitations of computer codes used.

12.23. The communication of Level 2 PSA results should consider the target audience and consequently the amount of information and data provided. The different reports, as stated in para 12.13, are suitable sources of information that could be communicated depending on the target audience.

12.24. Considering that reports on Level 2 PSA study might provide sensitive information, security considerations should be taken for the communication of those reports.

## 13. LEVEL 2 PSA FOR A SPENT FUEL POOL

13.1. Interest in the risk related to accidents occurring in the spent fuel pool has increased in Member States after the Fukushima Daiichi nuclear accident. A Level 2 PSA for the spent fuel pool might not be necessary, given the generally low frequency, long time frames for proceeding to fuel damage and the potential limited mitigation capabilities once fuel damage has occurred in the spent fuel pool. Depending on the pool's location and plant design specifics, available capabilities to prevent off-site releases in case of fuel damage at a spent fuel pool could be limited. In particular, para. 6.68 of SSR-2/1 (Rev. 1) [3] states (footnote omitted):

> "For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated' and so as to avoid high radiation fields on the site."

13.2. This section focuses its recommendations for the development of Level 2 PSA when the spent fuel pool is located inside a building capable to ensure the confinement function[29] in severe accident conditions. If not, one practice has been to consider in Level 2 PSA that accidents involving damage of fuel stored in the spent fuel pool lead directly to large radioactive releases. A complement to this practice is to proceed with an analysis aiming at substantiating the capabilities for crediting some fission product retention in buildings or water sources in severe accident conditions.

13.3. In principle, the Level 2 PSA for the spent fuel pool is based on the same methodology as Level 2 PSA for the reactor core outlined in Sections 5-11. Accordingly, the general process for conducting Level 2 PSA for the reactor core can be adapted for the spent fuel pool, considering the specific aspects addressed in this Section.

13.4. The goal of performing a Level 2 PSA for the spent fuel pool should be clearly defined (usually this goal is similar to the goal of Level 2 PSA for the reactor). A spent fuel pool PSA study can be performed separate or combined with PSA for the reactor core, depending on the specific needs and applications for developing the Level 2 PSA. Further, the definition of the undesired end states in the Level 1 PSA and the location of the spent fuel pool (inside the containment, inside a robust building, or outside the containment) can determine the analysis needs for a Level 2 PSA for the spent fuel pool. For example, location of the pool determines whether an accident progression event tree is necessary to be developed or whether other factors that could reduce the source term could be taken into consideration (e.g. possibility to close the containment, (i.e. if the spent fuel pool is located inside the containment), availability of the ventilation system and of the spent fuel cooling system).

## INTERFACE WITH LEVEL 1 PSA FOR A SPENT FUEL POOL

13.5. Similar to carrying out a Level 2 PSA for a reactor, PDSs specific to the spent fuel pool can be considered in the development of a Level 2 PSA for the pool. Factors specific to spent fuel pool analysis include items which influence the accident progression and source term, such as: time since last core offload, the fuel loading in the pool (e.g. number of fuel assemblies, fuel burnup, fuel loading pattern), pool configuration (i.e. whether the spent fuel pool is isolated from or interconnected to the reactor, or to the spent fuel pools in other unit(s)), water inventory of the pool and water leak rate.

13.6. The undesired end states (e.g. uncovering of fuel stored in the spent fuel pool or during fuel handling, boiling of the pool water) defined in Level 1 PSA for the spent fuel pool, as described in SSG-3 (Rev.1) (paras 10.2-10.6) [4], should be addressed in Level 2 PSA.

13.7. If the spent fuel pool PSA and the reactor PSA are combined, the PDS should consider combined reactor and spent fuel pool PDS. Reactor accident sequences can impact the spent fuel pool, for example containment venting could accelerate boiling of the water in the SFP if the SFP is located inside the containment. In addition, reactor accident sequences that do not result in Level 1 reactor core damage events may impact the mitigation actions for the spent fuel pool accidents and may have to be considered for inclusion in the PDS.

[29] The confinement function is such that radioactive releases are limited.

## SEVERE ACCIDENTS PROGRESSION ANALYSIS OF FUEL STORED IN THE SPENT FUEL POOL

13.8. To support Level 2 PSA development (if such a development is needed, see para. 13.2), deterministic analyses should be performed to analyse the severe accident progression in the spent fuel pool using one or more computer codes capable of modelling the accident progression and severe accident phenomena in the spent fuel pool. Severe accident phenomena to consider in this analysis includes heat transfer within the pool, fuel racks, and to surrounding walls, fuel behaviour (fuel burnup, decay heat, cladding behaviour, etc.), fuel assembly and rack degradation (zirconium clad reaction and hydrogen generation, zirconium fire, and corium–concrete interaction), fission product transport. Such calculations should provide information on the fraction of the fuel assemblies that would be damaged depending on the fuel assemblies arrangement, burn-up and storage time in the spent fuel pool.

13.9. The boundary conditions should define the amount of fuel that is normally replaced during a refuelling outage and a full core unload (if it is prescribed by the operating procedures) to be considered in the calculations.

13.10. Severe accident management measures for prevention of fuel damage in the spent fuel pool as well as mitigation, such as reflooding of the spent fuel pool, or use of mitigative spray, should be considered in the severe accident progression.

13.11. The following accident progression parameters in the spent fuel pool should be considered: time to boiling; time to fuel uncovering; time to fuel damage; time to spent fuel structure breach; time to penetration of concrete around spent fuel pool (if credited); source term magnitude and timing.

13.12. Depending on the plant configuration (spent fuel pool in or outside the reactor containment building), severe accident analysis should consider the interactions between the reactor and the spent fuel pool: a reactor accident can have impact on or induce a spent fuel pool accident and vice versa. From this analysis, some additional accident scenarios (involving both reactor and spent fuel pool) could be built in the Level 2 PSA if not already considered in the Level 1 PSA, such as the following:

(a) Common events which have an impact on the reactor and the spent fuel pool simultaneously (e.g. station blackout) or resulting consequential failure during the accident progression that affects safety functions for the other radioactive source;
(b) Impact of the accident management strategies for the reactor core to the spent fuel pool (e.g. if the spent fuel pool is located inside the reactor containment building, actuation of the filtered containment venting system will lead to more intensive water boiling in the spent fuel pool);
(c) Hydrogen release that could result in deflagration/detonation events that fail structures or electrical/mechanical equipment;
(d) Fission product releases that inhibit or preclude access to the areas needed for local manual actions;
(e) Effects of heat radiation from the damaged spent fuel pool on the containment structure or steel liner;
(f) Failure of all installed equipment, which may force the operators to decide where to prioritize the use of any remaining non-permanent equipment.

13.13. If the spent fuel pool is located inside the reactor containment building, accident progression analysis should address the impact from combined reactor and spent fuel pool accident on conditions in the containment (e.g. pressure, temperature, corium spreading, inflammable gas).

13.14. Hydrogen generation should be considered for loss of heat removal scenarios which cause evaporation of the large amount of water from the spent fuel pool. For spent fuel pools located in the containment, the capacity of the existing severe accident management arrangements for the reactor core (e.g. passive autocatalytic recombiners, filtered containment venting systems) should be verified whether they are sufficient to cope with the hydrogen additionally generated from the damaged fuel stored in the spent fuel pool.

13.15. Generated hydrogen may leak into adjacent or connected rooms to the spent fuel pool. The retention of hydrogen in the rooms on the release path that might lead to combustion and damage to the rooms should be justified.

13.16. Under dry conditions in the spent fuel pool, the risk of zirconium fire (i.e. water cooled reactors) and its propagation should be considered.

13.17. In general, spent fuel pool criticality is not likely due to the amount of fissile material in the SFP, as well as its geometrical configuration and presence of neutron absorbing material. Nevertheless, the issues of criticality in SFP should be addressed in the Level 2 PSA documentation.

13.18. For a spent fuel pool located inside a robust building, the overpressure created by steam should be considered when crediting retention of fission products inside the building.

ANALYSIS OF ACCIDENTS DURING FUEL TRANSFER OPERATIONS BETWEEN THE REACTOR AND THE SPENT FUEL POOL

13.19. If not screened out, dedicated analysis to address in the Level 2 PSA the consequences of accidents during fuel transfer operations between the spent fuel pool and the reactor, should be considered. Typical accidents to be considered are related to fuel uncovering due to the loss of spent fuel cooling system caused, for example, by a station blackout or effects due to external hazards (e.g. a seismic event).

ACCIDENT PROGRESSION EVENT TREE FOR A SPENT FUEL POOL

13.20. If the spent fuel pool is located inside the containment, the accident progression event tree can be developed similarly as for the reactor Level 2 PSA. Nevertheless, the analyst should introduce in the accident progression event tree the dependencies between the reactor and the spent fuel pool (systems, human actions, containment response); in comparison with a stand-alone spent fuel pool Level 1 PSA, some additional scenarios of spent fuel pool accidents induced by a reactor core melt accident may be introduced. For spent fuel pool outside the containment the accident progression event trees for the spent fuel pool Level 2 PSA can be simple event trees depending on the extent to which mitigation strategies are credited.

13.21. The harsh environments (high temperature, humidity and radiation levels) expected to be present around the spent fuel pool in a Level 2 PSA should be carefully considered when crediting local operator actions in the Level 2 PSA.

13.22. The effect of these harsh environments (high temperature, humidity and radiation levels) expected to be present around the SFP in a Level 2 PSA on the survivability of mitigative equipment, such as hoses, fittings and nozzles present around the spent fuel pool, as well as availability of the instrumentation system (e.g. spent fuel pool water level) should also be considered in a Level 2 PSA.

SOURCE TERM AND RELEASE CATEGORIES FOR A SPENT FUEL POOL

13.23. The source term calculations for a spent fuel pool Level 2 PSA could be performed similarly to those performed for the reactor Level 2 PSA.

13.24. Release paths from the spent fuel pool to the environment to be considered depend on the location of the pool inside or outside the containment. If the pool is located inside the containment, the potential release paths to the environment are almost the same as for reactor core melt accidents. Additional release paths should be reviewed, e.g. after penetrating the concrete wall or bottom of the spent fuel pool, the molten debris could come into contact with the containment wall and penetrate it leading to another type of late containment failure. If the pool is located outside the containment, the potential release paths to the environment depend on plant specific design, such as ventilation systems, building doors, roof under thermal impact, and size of rooms on the path.

13.25. Deposition of aerosols in the building on the release paths would mitigate the environmental impact and should be considered.

13.26. A dedicated source term analysis should be performed for the spent fuel pool and for accidents involving fuel assemblies transfer, based on the age distribution of the fuel elements. The core inventory of the spent fuel pool at potential accident times should be analysed considering the history of refuelling and the subsequent mixture of newer and older fuel elements.

13.27. Similar to the reactor PSA, release categories can be defined for the spent fuel pool to group similar accident sequences based on magnitude and timing of the release and calculate an associated frequency of release.

QUANTIFICATION AND ANALYSIS OF RESULTS FOR A SPENT FUEL POOL

13.28. All recommendations provided in Section 11 are applicable to the spent fuel pool PSA. In addition, the PSA models for the reactor and for fuel in the spent fuel pool should be integrated to correctly model dependencies of the shared systems. This is particularly important for the initiating events affecting both reactor and spent fuel pool simultaneously and for further Level 2 PSA study (in particular for plants with the pool inside the containment).

## 14. LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.1. Paragraphs 14.2 to 14.31 aim at providing recommendations for the development of Level 2 PSA for sites where several units are located, given that national regulatory requirements compel such studies. The development of such Level 2 PSA is not yet a common practice among the Member States, but it can present an interest to capture some risks relevant to the whole site as well as dependencies among units from the Level 2 PSA perspective, if they were not already addressed in the development of the PSA model for each single unit. Therefore, the recommendations in this Section are intended to harmonize the development of such studies among the Member States. More information on Member States' experience, practical case studies and guidance on PSA for multiple unit nuclear power plants are provided in Ref. [65].

## OBJECTIVES OF LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.2. The objective of the development of the Level 2 PSA for multiple unit nuclear power plants is to complement the Level 2 PSA of the single units on the site related to topics which are not fully addressed.

## SCOPE OF LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.3. In the scope of the Level 2 PSA for multiple unit nuclear power plants, the analyst should identify the list of topics which are not fully addressed in the single unit Level 2 PSA. Among the topics of interest might be:

(a) Hazards affecting the whole site, especially the implications on the severe accident mitigation system, equipment reliability and the human resources;
(b) Correlated or shared SSCs and resources among different units;
(c) Impact of consequences induced by a unit with a severe accident on the other units (e.g. fuel melt accidents happening in another unit).

14.4. The analyst should justify the introduction or not of those topics in the multi-unit Level 2 PSA with a screening process. A Level 2 PSA for a multiple unit nuclear power plant should consider the reactor units (such as power and/or research reactors) and the spent fuel pools on the site.

14.5. The selection of topics of interest should be such that their treatment will not induce excessive complexity in the development of the Level 2 PSA for multiple unit nuclear power plants.

## PREREQUISITES OF LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.6. Recommendations provided in paras 4-4.18 related to plant familiarization for the development of a single unit Level 2 PSA are also applicable as prerequisites for the development of the Level 2 PSA for multiple unit nuclear power plants.

14.7. The availability of a Level 1 PSA model for a multiple unit site is a prerequisite for the development of Level 2 PSA for a multiple unit site.

## RISK METRICS FOR LEVEL 2 PSA FOR A MULTIPLE UNIT NUCLEAR POWER PLANT

14.8. Traditional risk metrics used in PSA for a single unit site (e.g. large release frequency) could be used as far as possible in order to express the risk profile in the context of multiple unit nuclear power plants for corresponding decision-making (see paras 2.17-2.19). When relevant, these traditional risk metrics could be adapted in specific multi-unit risk metrics such as conditional probability of large releases from several reactors knowing large releases from one reactor of a unit on a multi-unit site.

## INTERFACE BETWEEN LEVEL 1 PSA AND LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.9. The Level 2 PSA for a multiple unit nuclear power plant begins when one unit is affected by fuel damage. The Level 1–Level 2 PSA interface should transfer the information on those units considered in the Level 2 PSA.

14.10. In principle, the PDS methodology for single unit PSA can be applied to PSA for multiple unit nuclear power plants as described in Section 5. The attributes for PDSs have to be adapted to both represent all units and limit the complexity of the model. The extent of the adaptation of the PDSs should depend on the identified topics of interest.

14.11. PDSs for accidents involving multiple units should group accident sequences coming from Level 1 PSA for multiple unit nuclear power plants that are equivalent for the risk of release considering the parallel evolution of all units on site (kinetics of accident progression and availability of mitigating systems).

## ACCIDENT PROGRESSION AND CONTAINMENT ANALYSIS IN LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.12. Existing studies for single unit Level 2 PSA should be used as the bases (described in Section 6) as far as possible for the accident progression and containment analysis in the context of a Level 2 PSA for a multiple unit nuclear power plant. Additional accident progression analyses may be needed depending on the differences of reactor technologies / designs on the site and the identified topics of interest.

14.13. The same general techniques and tools can be applied to perform the multiple unit accident progression analysis, with due consideration for the availability of mitigating systems, shared systems and the ability of operators to perform actions.

14.14. The major consideration which may differ from analysis for single units is the potential for correlated phenomenological factors to influence the accident progression. These factors include but are not limited to the various severe accident phenomena discussed in Sections 6, 7 and 8.

14.15. Ultimately, the way in which these factors combine in an event involving multiple units could influence the timing and magnitude of the generation of volatile fission products, containment failures and releases from containments.

## HUMAN AND EQUIPMENT RELIABILITY ANALYSIS IN LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.16. The recommendations provided in Section 8 should be considered applicable for Level 2 PSA for multiple unit nuclear power plants with regard to human and equipment reliability analysis, since the operator actions undertaken and credited within single unit PSA still apply to a multiple unit assessment. Specific aspects of Level 2 PSA for multiple units (e.g. recovery post-core damage, ensuring containment performance) necessitates a consideration of the impact due to the state of other units on site in terms of available resources and feasibility of specific Level 2 actions (discussed in Section 8).

14.17. For plants with multiple units, the interactions between the units (both positive and negative from risk point of view) should be considered in Level 1 PSA from the perspective of the unit under consideration and this applies to the shared systems between units and the degree of shared responsibility in the operator response to accidents.

14.18. The dominant consideration for human reliability assessment purposes here would be whether the broader organization of the site depends on common human teams to provide some of the responses necessary in accident mitigation (e.g. a single common fire brigade for the whole site). In such instances, consideration for the ability of those responses to succeed in parallel on multiple units should be considered.

14.19. Sites with multiple units of the same design are more likely to share common facilities such as a common main control room, power systems, switchyards and cooling water intakes. Multiple units with different designs might also share systems and teams. In the event of concurrent events at multiple units, the impact on operator actions may have implications for Level 1 PSA for multiple unit sites that should be considered for Level 2 PSA for multiple unit sites.

ACCIDENT PROGRESSION EVENT TREE FOR LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.20. The development of an accident progression event tree for Level 2 PSA for multiple unit nuclear power plants should be based on the recommendations provided in Section 9, which apply generally to Level 2 PSA for multiple unit nuclear power plants. At the single unit level, these approaches continue to describe the progression of accidents through initiating event, system failures and operator intervention in the accident progression event trees, and the consideration of nodal questions which describe the challenges to, and integrity of containment in the accident progression event tree. More specific discussion of accident progression event trees is provided in Section 9.

14.21. The same general theme of the degree of interconnectedness between units applies here. Consideration for the sharing of SSCs between units as well as the potential consequences of one unit on the successful operation, or mitigation of accident conditions on another unit should be made.

14.22. Since the number of units can add significant complexity and size to the accident progression event tree, consideration should be made to simplifying the single unit models before combination. Since each Level 1 sequence results in multiple Level 2 sequences by definition, it is prudent to simplify where possible. Methods to simplify the modelling could include but not be limited to the justified removal of low-risk initiating event contributors, focus first on those initiating events that could affect several units at the site at the same time (e.g. total loss of external power supply, total loss of ultimate heat sink, external flooding, earthquake), the grouping of similar Level 1 sequences under a single PDS and/or grouping release categories to capture the generic representation of an accident sequence.[30]

[30] For example, for a three unit site, having end states for one, two and three units resulting in releases, as opposed to U1, U2, U3, U1+U2, U1+U3, U2+U3, and U1+U2+U3.

## SOURCE TERM AND RELEASE CATEGORIES IN LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.23. The recommendations provided in Section 10 on source term analysis in single unit Level 2 PSA should also be considered for source term analysis in Level 2 PSA for multiple unit nuclear power plants, since they generally apply the same assessment of source terms, and the resulting release categories assignments. Much of the existing framework for release categories may be used as is, with consideration of where interconnectedness and dependencies for multiple unit Level 2 PSA may arise (as captured in the interface with multiple unit Level 1 PSA ).

14.24. In the analysis of source term and release categories in Level 2 PSA for multiple unit nuclear power plants, account should be taken of small and early releases from different sources at the site since they might become relevant when aggregated for the calculation of the large release or large early release risk metrics for the site.

14.25. It may be desirable to simplify the release categories once multiple units are considered to avoid an exponential increase in the potential combinations with limited benefits in terms of obtaining risk insights from the multiple unit Level 2 PSA.

14.26. It may be justifiable to further reduce the total number of release categories for Level 2 PSA for multiple units by consideration of the relative contribution of the individual release category on a specific basis, such as release magnitude, classes of release category in terms of their activity (i.e. Bq), frequency and/or timing. For example, if a particular binary combination is overwhelmingly driven by the contribution of one release category, engineering judgement may be applied to consider if additional granularity is needed for the purposes of Level 2 PSA for multiple unit nuclear power plants in comparison to a single unit analysis.

14.27. The addition of spent fuel pool accidents or shutdown states to the scope of a Level 2 PSA for multiple unit nuclear power plants may also further complicate the source terms analysis for multiple unit PSA due to the further increase in combinations of outcomes (even beyond those discussed in 13.28). By similar reasoning, screening and simplification of those outage states and radioactivity sources can be applied.

14.28. The aggregation and simplified combination of release categories for various facilities on a given site can be further assessed based on the commonality or differences in the definitions of releases from various facilities. While a reactor and spent fuel pool may have differing characteristics of core or fuel damage, it is easier to define releases from either facility as exceeding a defined release magnitude and facilitate easier binning of contributors to a single release category.

## QUANTIFICATION AND ANALYSIS OF RESULTS IN LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.29. The integration and quantification process for Level 2 PSA for multiple unit nuclear power plants should be based on the approach used in the single unit Level 2 PSA. In case of coupling PSA models from different units into a single PSA model, the major concern would be additional complexity from the additional event tree end states, release categories and combinations discussed above. It can be expected that quantification will involve additional consolidation and screening to include a manageable set of inputs for Level 2 scenarios that need to account for the effect of multiple units undergoing Level 1 and Level 2 aspects.

14.30. The treatment of sensitivity and uncertainties does not bear any major methodological differences to Level 2 PSA for multiple unit nuclear power plants, but it should be expected that a manageable addition for specific multiple unit impacts via sensitivity cases for phenomenology and modelling uncertainties is necessary to appropriately characterize the state of knowledge for multiple unit Level 2 PSA.

DOCUMENTATION OF THE ANALYSIS IN LEVEL 2 PSA FOR MULTIPLE UNIT NUCLEAR POWER PLANTS

14.31. The recommendations provided in Section 12 should be considered applicable to a Level 2 PSA for multiple unit nuclear power plants as well. There are no novel sections beyond the existing areas of documentation that need to be addressed. However, additional details and discussion regarding the characteristics of releases, the phenomenology which may arise due to multi-unit accidents and the resulting sensitivity and uncertainty in the source terms and release categorization should be well documented.

## 15. USE AND APPLICATIONS OF LEVEL 2 PSA

15.1. This Section provides recommendations on meeting Requirement 23 of GSR Part 4 (Rev.1) [2] on use of the safety assessment for Level 2 PSA. PSA has been applied in the design and operation of nuclear power plants in many Member States to complement results obtained by traditional methods of safety assessment. Many PSA applications use the results of Level 1 PSA (see SSG-3 (Rev. 1) [4]) and often also require Level 2 PSA results. The following list includes some successful examples of applications of Level 2 PSA; it should be noted that these applications of Level 2 PSA are not in use in every State:

(a) Comparison of results of the Level 2 PSA with probabilistic goals or criteria to determine if the overall level of safety of the plant is adequate;
(b) Evaluation of plant design to identify potential vulnerabilities in the mitigation of severe accidents;
(c) Development of severe accident management guidelines that can be applied following core damage;
(d) Use of the source terms to provide an input into the development of emergency preparedness and response arrangements;
(e) Use of the source terms and frequencies to determine off-site consequences (Level 3 PSA);
(f) Prioritization of research relating to severe accident issues;
(g) Use of a range of other PSA applications in combination with the Level 1 PSA results;
(h) Development of a list of severe accident scenarios to be addressed in the NPP design..

SCOPE AND LEVEL OF DETAIL OF LEVEL 2 PSA FOR APPLICATIONS

15.2. The scope and the level of detail of the Level 2 PSA should be consistent with its intended uses or applications, examples of which are described below. For example, the scope and the level of detail of a PSA that was intended to provide an estimate of the large release frequency or the large early release frequency and be used to provide insights into the potential failure modes of the containment will be different from the scope of a Level 2 PSA that was intended to provide an input into emergency preparedness and response or to a Level 3 PSA. In the calculation of large release frequencies or large early release frequencies, there is a need to identify accident sequences and their frequencies where the release would be categorized as 'large'. However, for the purposes of emergency preparedness and response the release characteristics (including the associated source terms) and, with limited extent, the frequencies associated to the occurrence of such releases, would need to be specified more accurately. For Level 3 PSA development, both the source terms and their frequencies would need to be specified more accurately. In addition, the level of detail of the PSA would need to be greater if it were intended to use the Level 2 PSA model in a risk monitor.

15.3. The scope of the Level 2 PSA, as stated in para. 2.7, should be commensurate with its intended uses and applications, and based on the equivalent scope of Level 1 PSA. A full scope of Level 2 PSA is most suitable for a large number of uses and applications, with due considerations given to the uncertainties on key parameters and limited strength of knowledge on some data and assumptions that could impact the PSA results and insights. Since the Level 2 PSA relies on the Level 1 PSA model, this should require that the Level 1 PSA:

(a) Includes an as comprehensive as possible set of internal initiating events, internal hazards, natural and human induced external hazards, and
(b) Addresses all plant operational states, including startup and operation at power, low power and all the modes that occur during plant shutdown and refuelling (if not screened out).

15.4. In any case, when the risk insights are to be derived from a Level 2 PSA that has a smaller scope than the full scope described in paragraph 15.3 (e.g. not all initiating events and hazards considered), this should be recognized in applying the insights from the PSA.

15.5. This scope will ensure that the insights from the PSA relating to the risk significance of accident sequences, SSCs, human errors, common cause failures, are derived from a comprehensive, integrated model of the plant. If the Level 2 PSA is based on a Level 1 PSA that has a more limited scope or details, these limitations need to be taken into account in the application of the Level 2 PSA.

## USE OF LEVEL 2 PSA THROUGHOUT THE LIFETIME OF THE PLANT

15.6. As recommended in paras 2.20 to 2.23, the Level 2 PSA used for any application should be actively maintained and regularly updated, taking into account changes in plant design and operational practices as well as feedback from experience and advances in technology that may compromise the validity of the PSA. For the Level 2 PSA, this updating needs to take account of changes in the provisions made and the guidance provided for severe accident management, updates to the severe accident analysis carried out to support the Level 2 PSA model and the results of research carried out that provide a better understanding of the phenomena that occur during a severe accident.

15.7. The Level 2 PSA should be used to provide one of the inputs into design evaluation throughout the lifetime of a nuclear power plant. It should be used during the design process for a new plant to determine whether adequate features for the mitigation of severe accidents are being incorporated into the design of the plant and this should be updated throughout the construction and operational stages of the lifetime of the plant.

15.8. The Level 2 PSA should also be used to provide an input into the development of the severe accident management guidelines, which should be available when the plant goes into operation.

## RISK INFORMED APPROACH TO LEVEL 2 PSA

15.9. The aim of applying a risk informed approach is to ensure that a balanced approach is taken when making decisions on safety issues by considering probabilistic risk insights with any other relevant factors in an integrated manner (see Refs [66], [67]).

15.10. In any of the applications of the Level 2 PSA described below, the insights from the PSA should be used as part of the process of risk informed decision making that takes account of all the relevant factors when making decisions on issues related to the prevention and mitigation of severe accidents at the plant:

(a) Any mandatory requirements that relate to the PSA application being addressed (which would typically include any legal requirements or regulations that need to be complied with);
(b) The insights from deterministic safety analysis;
(c) Any other applicable insights or information (such as include a cost–benefit analysis, remaining lifetime of the plant, inspection findings, operating experience, doses to workers that would arise in making necessary changes to the plant hardware, and environmental protection concerns).

## COMPARISON OF LEVEL 2 PSA WITH PROBABILISTIC SAFETY CRITERIA OR GOALS

15.11. The overall results of the Level 2 PSA should be compared with the probabilistic safety goals or criteria (if these have been specified, see paras 2.17-2.19). The aim should be to determine whether the risk criteria or goals have been met or whether additional features for prevention or mitigation of accidents need to be provided.

15.12. This comparison should take account of the results of the sensitivity analyses that have been carried out and the uncertainties inherent in the Level 2 PSA. The sensitivity analyses and the uncertainty analyses should be used to indicate the degree of confidence in meeting the criterion or target and the likelihood that it may be exceeded.

15.13. In 1999, probabilistic criteria were proposed by the International Nuclear Safety Advisory Group (INSAG) [7] for a large off-site release of radioactive material requiring a short-term off-site response[31]. Several States have also set similar numerical values which have generally been defined as objectives or targets (see Annex IV).

15.14. In addition, for future nuclear power plants, rather than defining probabilistic criteria, INSAG [7] has proposed that the objective should be "the practical elimination of accident sequences that could lead to large early radioactive release, whereas severe accidents that could imply late containment failure would be considered in the design process with realistic assumptions and best estimate analysis so that their consequences would necessitate only protective measures limited in area and in time."

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[31] According to Ref. [7], the objective for large off-site releases requiring short term off-site response is $1 \times 10^{-5}$ per reactor-year for existing plants.

LEVEL 2 PSA FOR DESIGN EVALUATION

15.15. The Level 2 PSA should be used to carry out a safety evaluation of the plant design. The aim should be to gain insights into how severe accidents progress, identify plant specific vulnerabilities and provide an input into the consideration of whether improvements need to be made to the design of the plant.

**Identification of plant vulnerabilities**

15.16. The use of Level 2 PSA for design evaluation is very similar to that for Level 1 PSA, as described in SSG-3 (Rev. 1) [4]. As well as calculating the overall value of the large release frequency or large early release frequency, the computer codes used to develop the Level 2 PSA model and to quantify it provide a range of other information including:

(a) The frequency of each of the release categories;
(b) The possible combinations of failures (cutsets) that contribute to each of the release categories;
(c) The importance functions for systems, components and other basic events included in the PSA model. (This will depend on the computer code used for the development of the Level 2 PSA but could include the Fussell-Vesely importance, the risk achievement worth, the risk reduction worth, and the Birnbaum importance).

15.17. The information provided by the Level 2 PSA should be used to identify weaknesses in the features provided for the prevention and mitigation of severe accidents. This information could include:

(a) The significant failure modes of the primary circuit and the containment;
(b) The dominant phenomena that lead to (early or late) containment failure;
(c) The SSCs that have the highest importance for large release frequency or large early release frequency.

15.18. Consideration should be given to making improvements to the features provided for the prevention or mitigation of severe accidents in order to reduce contributions to the overall risk of sequenceswith the highest risk significance.

15.19. The improvements considered should include the provision of additional protective systems and features for mitigating the consequences of the severe accident. This could involve incorporating such additional protective systems and features into a new design or backfitting them into an existing plant.

15.20. The results of the Level 2 PSA should be used as a resource for determining whether adequate provisions for defence in depth have been made. For example, the PSA could provide a basis for determining whether severe accident management measures and guidelines fully address the fourth level of defence in depth as defined in SSR-2/1 (Rev. 1) [3].

**Comparison of design options**

15.21. When design improvements are being considered with regard to severe accident management measures, a range of options are often available. The Level 2 PSA may be used to provide an input into the comparison of these options according to paras 2.24-2.35.

15.22. The Level 2 PSA should be used to compare the benefits in terms of risk reduction and balance of the design from the incorporation of these additional systems and features. The way that this is done depends on the complexity of the design options being considered but could range from the production of a revised PSA model to post-processing the cutsets to take account of simpler changes and even to carrying out sensitivity studies that relate to the design options. In doing this, it needs to be recognized that a design change may impact a whole sequence of events modelled in the accident progression event tree, or even change the basis for evaluation of some nodes of the accident progression event tree. A design change might also affect the Level 1 PSA. Competing impacts need to be recognized and taken into account in the evaluation of the design change. As an example, a modification to the spray system may benefit the control of steam pressurization but may have the potential to lead to combustible conditions in some time frames, or even lead to concerns about containment underpressure.

USE IN DEVELOPMENT OF SEVERE ACCIDENT MANAGEMENT GUIDELINES

15.23. The Level 2 PSA should be used as a basis for the evaluation of the measures in place and the actions that can be carried out to mitigate the effects of a severe accident after core damage has occurred. The aim of mitigatory measures and actions should be to arrest the progression of the severe accident or mitigate its consequences by preventing the accident from leading to failure of the containment or the reactor pressure vessel (for in-vessel melt retention strategy) and controlling the transport and release of radioactive material with the aim of minimizing off-site consequences. Examples of mitigatory actions that could be carried out for pressurized water reactors include:

(a) Opening the pressurizer relief valves in order to reduce the reactor coolant system pressure and so avoid molten core material being ejected from the reactor pressure vessel under high pressure;
(b) Adding water to the containment by any available means after the molten core has exited from the reactor coolant system so as to provide a cooling mechanism.

15.24. The results of the Level 2 PSA should be used to determine the effectiveness of the severe accident management measures that are described in the severe accident management guidelines or procedures, whether they have been specified using the Level 2 PSA or by any other method.

15.25. In developing a Level 2 PSA, it should be recognized the associated uncertainties to the probability for phenomena that occur in the course of a severe accident and their interrelations, so that an accident management measure that is aimed at mitigating a particular phenomenon might make another phenomenon more likely. Examples of this for pressurized water reactors include the following:

(a) Depressurization of the primary circuit may prevent high pressure melt ejection but might increase the probability of an in-vessel steam explosion;
(b) Introducing water into the containment may provide a cooling medium for molten core material after it has come out of the reactor pressure vessel but might increase the probability of an ex-vessel steam explosion;
(c) Operation of the containment sprays may provide a means of removing heat and radioactive material from the containment atmosphere but might increase the flammability of the containment atmosphere by condensing steam.

15.26. These interdependencies between the various phenomena that can occur during a severe accident should be identified using the Level 2 PSA and should be taken into account in the development of the severe accident management guidelines. Updates of the Level 2 PSA and updates of the severe accident management guidelines should be performed in an iterative manner to facilitate the progressive optimization of the severe accident management guidelines.

PRIORITIZATION OF RESEARCH ACTIVITIES ON SEVERE ACCIDENTS

15.27. Level 2 PSA models the complicated and highly interrelated phenomena that occur after a severe accident. Although there has been a considerable amount of research into these phenomena, there is still a lack of knowledge in some areas that leads to a significant level of uncertainty in the predictions of the Level 2 PSA.

15.28. The Level 2 PSA should be used to provide a basis for the identification and prioritization of research activities. Such research activities should focus on identifying what areas of research can contribute most to improving knowledge aiming at reducing the uncertainty in the highest risk significance parameters or phenomena.

INPUT FOR LEVEL 3 PSA

15.29. The source terms and frequencies derived in the Level 2 PSA can be used as the starting point for determining the off-site consequences that can result from releases of radioactive material from the plant. Such off-site consequences include health effects to members of the public and a range of consequences, including contamination of land, water and food, evacuation, and permanent relocation.

15.30. If a Level 2 PSA is interfaced with a Level 3 PSA, consideration should be given to revisiting the release category definitions to ensure that all the information needed for the Level 3 PSA is available in the Level 2 PSA end states.

15.31. The source terms and frequencies derived in the Level 2 PSA should be used as the starting point for the Level 3 PSA carried out to address the off-site consequences that could arise from a severe accident at the plant. The scope of the Level 2 PSA to be used for this purpose should include a detailed model of the transport of radioactive material and its release from the plant.

EMERGENCY PREPAREDNESS

15.32. The source terms and, to some limited extent, their frequencies derived in the Level 2 PSA, along with projections of the off-site dose as a function of distance, should be used as inputs into the development of off-site emergency preparedness and response arrangements. One or more reference accidents can be defined and used in this process.

15.33. For a Level 2 PSA that is to be used for emergency preparedness and response, the releases considered should be accurately specified in terms of isotopic composition, amount and timing of radioactive material released (i.e. source terms), as well as in terms of relevant additional attributes (see TABLE 7 in Section 10).

15.34. Requirement 4 of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [68] states that "**The government shall ensure that a hazard assessment is performed to provide a basis for a graded approach**

**in preparedness and response for a nuclear or radiological emergency.**” With a view to meeting this requirement and Requirement 23 of GSR Part 4 (Rev.1) [2] on use of the safety assessment as part of the integrated risk informed approach, the source terms and, to some limited extent, their frequencies derived in the Level 2 PSA can be used as an input to determine the extent of the emergency planning zones and emergency planning distances.

OTHER PSA APPLICATIONS

15.35. The Level 2 PSA should be used in combination with the Level 1 PSA results for a number of applications, as described in SSG-3 (Rev. 1) [4] for the Level 1 PSA. The use of Level 1 and Level 2 PSAs in combination will provide additional insights to those obtained solely from the Level 1 PSA, since the relative importance of SSCs is normally different for Level 2 PSA results, such as large release frequency or large early release frequency, than for Level 1 PSA results, such as core damage frequency.

# APPENDIX I. CONSIDERATIONS FOR HUMAN RELIABILITY ASSESSMENT IN A LEVEL 2 PSA

AI-1. Significant differences in context exist for the nuclear plant operators performing tasks before and after core damage to protect the plant and public safety. These differences significantly impact the factors to be considered in performing human reliability assessment for a Level 2 PSA. Example factors include command-and-control, coordination between the organizations involved to respond to the event, the training of the nuclear plant staff and the other organizations in responding to the events, the specifics of procedural instructions, the predictability of scenario progression, the degraded plant instrumentation, and the environmental impacts on task performance. The human reliability assessment community has accumulated a tremendous amount of experience in performing human reliability assessment for Level 1 PSA but much less for Level 2 PSA. This appendix discusses the above context differences and their implications in performing human reliability assessment in Level 2 PSA.

AI-2. When core damage is imminent or occurring, the operator would enter the severe accident management guidelines to mitigate the event. Significant changes to the command-and-control occur after entering the severe accident management guidelines to various degrees depending on the organization. These changes include the transfer of decision-making authority from the main control room to the technical support centre (or at least the inclusion of the technical support centre in the decision-making process), the supersedence of the emergency operating procedures with the severe accident management guidelines and a shift in event response focus from preventing core damage to preventing and reducing the release of radioactivity to the environment. In this new command-and-control structure, the technical support centre may decide or propose the mitigative strategies. Operators in the main control room implement the mitigative strategies and coordinate their implementation. Depending on the organization, after entering severe accident management guidelines, this practice may differ from Level 1 PSA, where operators in the main control room make almost all decisions, even if in some organizations, the technical support centre may also recommend specific actions in accident conditions.

AI-3. Multiple organizations may participate in responding to severe accidents. For example, the local fire brigade and companies with contracts with the plant for event response could come on-site to support event mitigation. The government and police could evacuate residents. Ambulances could come on-site to pick up injured workers. These organizations that do not conduct routine emergency response exercises with the plant could have issues in communication and coordination. Matters such as whether the contact information and contracts are up to date, whether the communication equipment is compatible, and whether the line of command is clear might have a significant effect on human reliability. For example, the technical support centre might delay in commanding the performance of controlled radioactivity releases if the line of command is not clear, and, as a result, the radioactivity release could impact evacuation.

AI-4. Most nuclear plants routinely use plant-specific, full-scope plant simulators to train their operators to prevent core damage. Most simulators cannot simulate prolonged post core damage phenomena. As a result, operator training on Level 2 scenarios mainly relies on classroom training that provides only high-level guidance. Operator reliability in knowing the plant status can be significantly affected if instrumentation is not available or not reliable. For example, if core crust shields the molten core from cooling water, the instrumentation may not provide information for the operator to evaluate the effectiveness of the cooling approach, or whether the reactor vessel has remained intact. In general, severe accident mitigation needs the implementation of specific sensors, qualified to severe accident conditions, in order to improve success of mitigation actions. Because of the significant uncertainty in scenario progression, the severe accident management guidelines might be written such that operators can more easily deviate from them than from the emergency operating procedures. In this case, the practice could be prone to errors of commission. The modelling of errors of commission therefore deserves careful consideration.

AI-5. Environmental factors such as high temperatures, high levels of radiation, seismic aftershocks, and blockage of equipment transportation can affect task performance, as shown at the Fukushima Daiichi and Daini nuclear power plants after the earthquake and tsunami event in 2011. After the Fukushima Daiichi accident, many nuclear power plants began to use non-permanent equipment stored outside the site to mitigate similar events. The transportation and operation of such non-permanent equipment takes place in an open environment that is susceptible to the impacts of adverse conditions. In addition, an adverse environment could affect the instrumentation availability and accuracy needed for operators' decisions.

AI-6. The above discusses the factors in severe accidents that affect the operators' cognitive functions in detecting plant information, understanding the plant status, making event mitigation decisions, and implementing event mitigation strategies. In addition, there are two significant differences between human reliability assessment practices in Level 2 PSA and Level 1 PSA. The first difference is about repairing unavailable components. Both emergency operating procedures and severe accident management guidelines may instruct the operators to repair unavailable components for event mitigation. While repair actions are hardly credited for Level 1 PSA, the long time window makes repairing components an option for Level 2 PSA. Operator interventions to mitigate severe accidents could still be needed even a few days after the core damage. The human reliability assessment community would need more data to credibly assess the probabilities of repairing unavailable components in Level 2 PSA. The second difference is about modelling task dependency. For example, a common practice of Level 2 PSA is that the sequences start with the plant damage state. Each plant damage state consists of many Level 1 scenarios. This practice simplifies the modelling of Level 2 PSA but it may have an influence on performing detailed human reliability assessment. Starting a scenario from a plant damage state instead of an initiating event, depending on the level of detail of the plant damage state defined, important scenario-specific information might not be carried to Level 2 PSA for detailed human reliability assessment. The missing information, such as initiating events, failures of the SSCs, availability of electric power, and previous operator errors, may be important for dependency analysis. A practice to reduce the effects on PSA results is to analyse the dominant minimal cutsets specific to the plant damage state to identify their representative event sequences from the initiating events to model dependency effects.

AI-7. A few human reliability assessment methods and processes have been developed with Level 2 PSA in their scopes. These include the expanded Human and Organizational Reliability Analysis in Accident Management (HORAAM) method [69] and [70], HAMSTER [71] MERMOS method [72], Integrated Human Event Analysis for Event and Condition Assessments (IDHEAS-ECA) method [73], and expert judgment using human reliability assessment process [74]. In addition, guidance [75] on applying the existing human reliability assessment methods to assess the human reliability of performing tasks in extreme conditions is available.

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## ANNEX I COMPUTER CODES FOR SIMULATION OF SEVERE ACCIDENTS FOR WATER COOLED REACTORS

I-1. Severe accident phenomena are complex and have many interdependencies which can be realistically examined using complex computer codes. This annex provides insights into the type of code typically used in Level 2 PSAs and a brief description of their areas of application.

GENERAL DESCRIPTION OF COMPUTER CODES

### Types of code

I-2. The codes that model the physical response of the core, the reactor coolant system and the containment to severe accidents can be divided into three types according to their capabilities and intended use:

(1) *Mechanistic* codes, also named as *'Detailed'* codes, calculate governing phenomena with best estimate models based on first principles, with computational resources being of secondary importance. Mechanistic codes are used typically in research to design and analyse severe accident experiments and to simulate as accurate as possible the phenomena and the behaviour of the nuclear power plant SSCs during the severe accident. Once validated against appropriate experimental conditions and tests (i.e. integral or simplified tests), they are also used to establish benchmarks for integral codes or to define simplified models in integral codes. Codes of this type span a wide range of technical disciplines, from the behaviour of damaged fuel to the release of radioactive material, and from transport to hydrogen mixing and combustion processes. Examples of codes in each of these areas are given in para. I-9.

(2) *Integral* codes, which are designed for routine application in PSA, generally use simplified models of some phenomena so that calculations can be completed relatively quickly (within hours or at most a few days with the current computing technology). As they are relatively fast running, these codes can be used to evaluate plant response to many different accident sequences or can be run several times for the same accident sequence to support sensitivity and uncertainty analysis. To ensure that the overall execution time of the code is reasonable, the modelling approach to some phenomena is simpler than the approaches used in mechanistic codes. The processes governing fuel damage and melting are offered as an example of the sort of simplification used. In a mechanistic code, models might be used to evaluate explicitly the individual effects of several damage mechanisms within fuel rods, including swelling of fuel pellets and 'foaming' due to the expansion of fission product gases, thermomechanical interactions between the swollen fuel pellet and bounding clad, local ballooning at weak points in the clad, changes in material composition and properties associated with formation of eutectic mixtures, material liquefaction and candling, etc. This same process might be treated in a simpler and composite manner in integral codes. For example, clad 'failure' (i.e., release of the gap inventory of radionuclides) might be represented by specifying an effective clad failure temperature, while the effect of eutectic formation on liquefaction properties of the fuel might simply be represented by reducing the effective 'melting temperature' of the fuel. The extent to which such simplifications properly reflect important characteristics of the actual governing phenomena is determined by comparison of the calculated results with experimental data and with the results of parallel calculations performed with mechanistic codes. Examples of such comparisons are found in Refs [I-1] and [I-2].

(3) *Dedicated* codes and algorithms, also named '*Fast running'* codes, provide rough estimates of parameters for specific PSA applications, such as estimation of the radiological source term [I-3] or of containment loads accompanying high pressure melt ejection [I-4]. Such tools are generally used to establish the primary technical basis when more runs are needed than can be reasonably handled, even by contemporary PSA codes. Dedicated codes are based on simple parametric models that interpolate between fixed points, for which calculations with a more complicated code have already been performed, to determine the values of the parameters. The use of such codes is reasonable for the generation of uncertainty values, but it is important to take into account that the parameters used in the codes, as well as the results produced by them, have to be calibrated by more detailed calculations or experimental data. Examples of those codes is provided in para I-10.

I-3. In the past, an approach was used where separate codes, each dealing with a particular phase or aspect of severe accident behaviour, were coupled in a suite, with some interfacing facility for the transfer of information between the codes. However, for routine PSA application, it is desirable to have automatic transfer of information between the elements of a code suite as manual transfer is slow and can also lead to the introduction of errors. A more integrated and modular approach has tended to be adopted in the newer generation of severe accident codes. As such, integral codes are able to model the feedback between different phenomena that could be missed in only the execution of a specific mechanistic model.

**Validation status of a code**

I-4. Verification and validation of computer codes are crucial mechanisms that enhance confidence in their application. Achieving a state with severe accident codes that could reasonably be termed validation is very difficult. However, the extreme conditions that occur in a severe accident and the scale of the physical geometry are difficult to realize in experiments. The process of validation, in general, comprises a validation matrix involving many simulations. Care needs to be taken with code validations that have been achieved by varying the values of user supplied parameters until a reasonable fit to experimental data is achieved. At best, this is an indirect experimental measurement of the parameter values and not an independent validation of the code.

**Use of the codes**

I-5. Deterministic accident analysis codes need to be designed so that a Level 2 PSA analyst having a good degree of familiarity with general accident phenomena can run them reliably without needing to have the same detailed knowledge as a specialist using a mechanistic code dealing with a particular phenomenon or a phase of a severe accident. However, it is essential that the analyst has a good working knowledge of the reactor systems. In order for the code calculations to be meaningfully incorporated into the framework of a Level 2 PSA, the analyst will need to have a reasonable knowledge of the following:

(a) The phenomena addressed in a code and their modelling approach and limitations;
(b) The meaning of the input variables and the range of validity for the selected code;
(c) The meaning of the output variables.

I-6. Given the complexity of these issues, the code cannot simply be treated as a 'black box'. The user will need to have a sound knowledge of the strengths and limitations of the code, which may not be used out of the range of situations and conditions for which it has been designed.

EXAMPLES OF INTEGRAL CODES FOR SEVERE ACCIDENT ANALYSIS

I-7. This Section provides a brief description of some specific codes currently in use for Level 2 PSAs, which deal with most or all of the phenomena shown in Fig. I–1. A list of major mechanistic codes is also included.

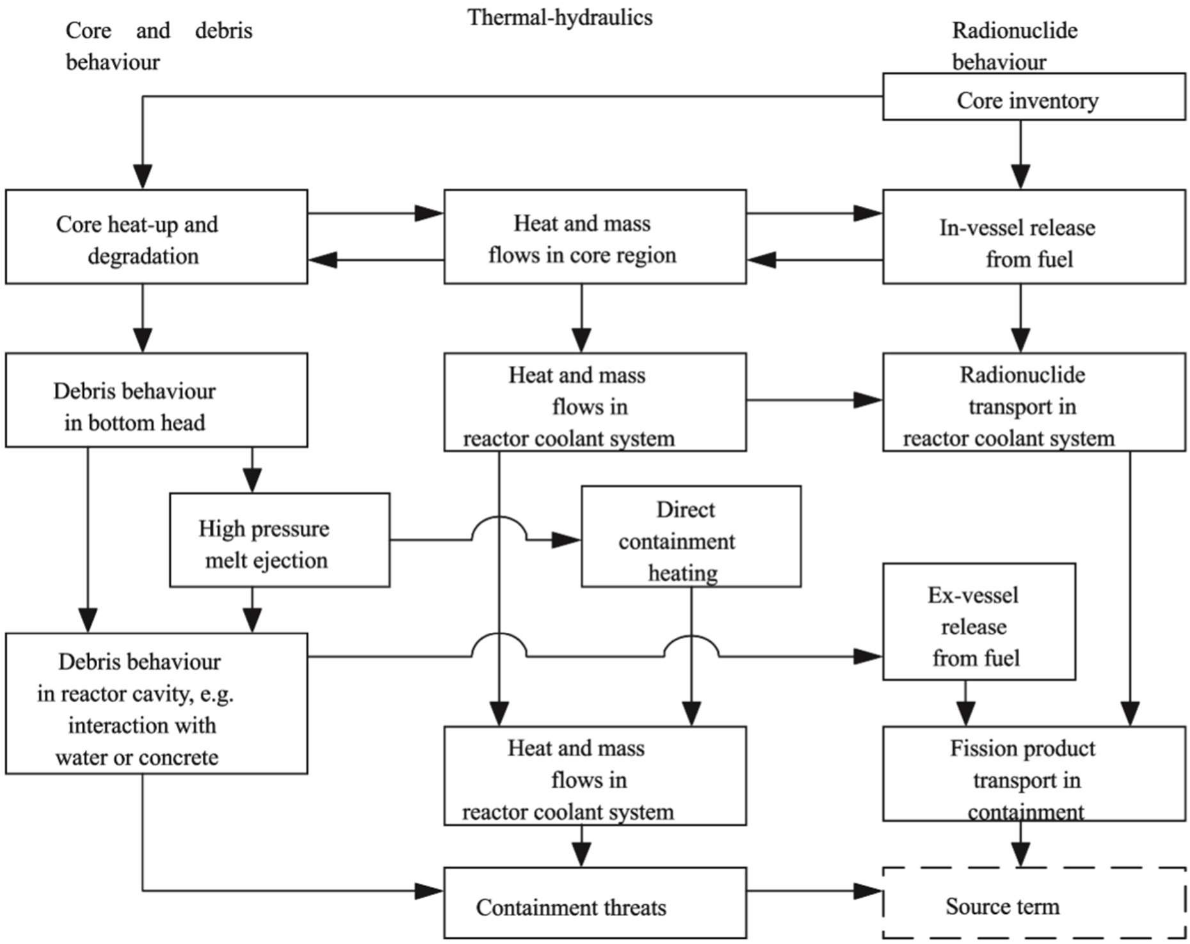


*FIG. I–1. General form of severe accident codes for light water reactors.*

**Integral codes**

I-8. Integral codes model the physical response of the entire plant to postulated severe accidents from the initiating event through to the release of radioactive material to the environment. The range of phenomena and processes modelled by such codes includes:

(a) Thermohydraulic processes in the reactor coolant system, the containment structure and/or the SSCs ensuring the confinement function;
(b) Degradation of core cooling, fuel heat-up, cladding oxidation, fuel degradation (loss of fuel geometry) and melting and relocation of core material;
(c) Heat-up of the reactor pressure vessel lower head from relocated fuel material and the thermal and mechanical loading and failure of the reactor pressure vessel lower head;
(d) Transfer of core material from the reactor pressure vessel to the containment 'cavity';

(e) Thermochemical interactions between molten core debris and concrete on the containment floor and resulting generation of aerosols;
(f) In-vessel and ex-vessel production of combustible gases (e.g. hydrogen and carbon monoxide), transport and combustion;
(g) Radioactive material release (aerosol and vapour), transport and deposition;
(h) Behaviour of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere, such as particle agglomeration and gravitational settling;
(i) Impact of engineered safety features on thermohydraulic and radionuclide behaviour.

**Mechanistic codes**

I-9. The level of detail examined by these codes generally exceeds that necessary for most Level 2 PSAs. Nevertheless, their application is occasionally necessary under special circumstances, such as when particular issues are unusually important to severe accident behaviour in a unique plant design.

**Dedicated codes**

I-10. Depending on the objective, dedicated codes could be based on a fast-running algorithm or a complex calculation to describe the single phenomenon under study. They can be used for different reactor technologies such as water cooled reactors and high temperature gas cooled reactors [I-35] and [I-36]

PSA COMPUTER CODES

I-11. Codes for simulation of fault trees and event trees and other simulation codes typically used for Level 1 PSAs are often used for Level 2 PSAs. In many cases, such codes have been adapted or enhanced to address certain unique requirements of Level 2 PSA applications, such as the solution of logic models with large event probabilities, and enhanced capabilities or more diverse methods for addressing uncertainties. For separated Level 1 - Level 2 PSA modelling, there are several specific Level 2 PSA codes that are designed to take care of the modelling of the Level 2 part (accident progression after core damage). Such codes have special features, e.g. user defined functions that calculate the severe accident event probabilities. A compilation of computer codes for Level 1 PSA and Level 2 PSA is provided in Ref. [I-37]. Codes that have been specifically developed for accident progression event tree analysis are generally very well qualified for phenomenological issues in Level 2 PSA but may have to be adapted to model the behaviour of systems.

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## ANNEX II SAMPLE OF PLAN OF ACTIVITIES AND OUTLINE OF DOCUMENTATION FOR A LEVEL 2 PSA STUDY

II-1. Given the great number of uncertainties associated with the performance of Level 2 PSA, the standardization of the outline of documentation for presenting Level 2 PSA studies allows better peer review process. An example of outline documentation for the summary report and the main report are presented below.

<u>SAMPLE CONTENTS OF THE SUMMARY REPORT</u>

S1. Introduction
S2. Overview of the objectives and motivation for the study
S3. Overview of the approach
S4. Results of containment failure modes and likelihoods
S5. Radiological source terms and their frequencies (complementary cumulative distribution functions)
S6. Summary of plant vulnerabilities to severe accidents, interpretation of results
S7. Conclusions and recommendations
S8. Possible risk reduction measures
S9. Organization of the main report

<u>SAMPLE CONTENTS OF THE MAIN REPORT</u>

M1. Introduction
- M1.1 Background
- M1.2 Objectives
- M1.3 Scope of the study
- M1.4 Project organization and management
- M1.5 Composition of the study team
- M1.6 Overview of the approach
- M1.7 Structure of the report

M2. Description of the design of the plant and the containment
- M2.1 Plant and containment design features affecting severe accidents
- M2.2 Operational characteristics
- M2.3 Description of plant modifications and containment system modifications (if any)

M3. Interface to Level 1 PSA
- M3.1 Grouping of accident sequences and specification of attributes
- M3.2 PDSs for internal initiating events and associated uncertainties
- M3.3 PDSs for external initiating events and associated uncertainties
- M3.4 PDSs for other power states and associated uncertainties

M4. Analysis of the containment's structural performance
- M4.1 Description of the structural design and failure modes of the containment
- M4.2 Approach for structural analysis
- M4.3 Structural response and fragility results
- M4.4 Summary of uncertainties and/or fragility curves for containment performance
- M4.5 Impact of external events

M5. Accident progression and containment analysis
- M5.1 Severe accident progression analysis

M5.1.1 Scope of analysis
M5.1.2 Method of analysis (codes, models, etc.)
M5.1.3 Summary of point estimate results for PDSs analysed
M5.2 Accident progression event trees
M5.2.1 Accident progression event tree structure
M5.2.2 Operating procedures and recovery
M5.2.3 Accident progression event tree quantification process
M5.2.4 Binning of Accident progression event tree end states
M5.2.5 Treatment of uncertainties
M5.2.6 Results
M5.2.6.1 Point estimate C matrix
M5.2.6.2 Uncertainties in failure probabilities
M5.2.6.3 Interpretation of results
M6. Accident source terms
M6.1 Grouping of radioactive materials
M6.2 Method of analysis (codes, models, etc.)
M6.3 Summary of point estimate results for PDSs analysed
M6.4 Treatment of uncertainties
M6.5 Results
M6.5.1 Point estimate source term characteristics
M6.5.2 Uncertainties in source term characteristics
M6.5.3 Interpretation of results
M7. Sensitivity and importance analyses
M7.1 Identification of sensitivity issues
M7.2 Results of sensitivity analysis
M7.3 Importance ranking of issues, systems and components
M8. Conclusions
M8.1 Key insights on characteristics of severe accidents and containment response
M8.2 Design features and inherent mitigation benefits
M8.3 Conclusions relative to PSA objectives
A. Appendices
A1. Basis for containment structural fragilities
A2. Basis for accident progression event tree quantification
A3. Results of deterministic severe accident analyses
A3.1 Containment loads
A3.2 Accident source terms
A4. Basis for probability distribution and ranges of uncertain parameters
A5. Detailed results of uncertainty analysis and/or sensitivity analysis

II-2. The successful performance of Level 2 PSA needs careful planning of relevant activities as part of the project management. In addition, the performance of planned activities for Level 2 PSA depends on the level of expertise and preparation of the team. The following table shows an example of planning of Level 2 PSA related activities taking into consideration the participation of a well-trained Level 2 PSA team.

TABLE II- 1 Example of plan for performance of Level 2 PSA

| Activities | Months | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | Planning | | | | Study performance | | | | | | | | | | | |
| | 1 | 2 | 3 | 4 | 1 | 2 | 3 | 4 | 5 | 6 | 7 | 8 | 9 | 10 | 11 | 12 |
| Management and organization | | | | | | | | | | | | | | | | |
| 1 Definition of objectives | ■ | | | | | | | | | | | | | | | |
| 2 Definition of scope | | ■ | | | | | | | | | | | | | | |
| 3 Project management plan | | | ■ | | | | | | | | | | | | | |
| 4 Selection of approach and establishment of procedures | | | | ■ | ■ | ■ | | | | | | | | | | |
| 5 Team organization | | | | ■ | | | | | | | | | | | | |
| 6 Team training | | | | | ■ | ■ | ■ | ■ | | | | | | | | |
| 7 Funding and scheduling | | | | ■ | | | | | | | | | | | | |
| 8 Setting a quality assurance process | | | | ■ | ■ | ■ | | | | | | | | | | |
| 9 Selection of internal peer reviewers | | | | ■ | ■ | | | | | | | | | | | |
| 10 Selection of external peer reviewers | | | | | ■ | ■ | | | | | | | | | | |
| Study performance | | | | | | | | | | | | | | | | |
| 11 Plant familiarization, identification of important design features | | | | | ■ | ■ | | | | | | | | | | |
| 12 Interface to Level 1 PSA, containment system modelling and sequence grouping | | | | | | ■ | ■ | ■ | | | | | | | | |
| 13 Accident progression and containment integrity analysis | | | | | | | | | | | | | | | | |
| a) Containment performance analysis | | | | | | ■ | ■ | ■ | | | | | | | | |
| b) Human and equipment reliability assessment | | | | | | ■ | ■ | ■ | | | | | | | | |
| c) Severe accident progression analysis | | | | | | | ■ | ■ | ■ | ■ | ■ | ■ | | | | |
| d) Development and quantification of accident progression event trees | | | | | | | | | | ■ | ■ | ■ | ■ | | | |
| e) Treatment of accident progression uncertainties | | | | | | | | | | | ■ | ■ | ■ | | | |
| f) Summary and interpretation of accident progression results | | | | | | | | | | | ■ | ■ | ■ | ■ | | |
| 14 Source term analysis | | | | | | | | | | | | | | | | |
| a) Grouping accident progression event trees end states into release categories | | | | | | | | | | | | ■ | ■ | | | |
| b) Source terms calculations | | | | | | | | ■ | ■ | ■ | ■ | ■ | | | | |
| c) Treatment of uncertainties in source terms | | | | | | | | | | | | ■ | ■ | | | |
| d) Summary and interpretation of results | | | | | | | | | | | ■ | ■ | ■ | ■ | | |
| Documentation of PSA results: Presentation and interpretation of results | | | | | | | | | | | | | | | | |
| 15 Integration, interpretation and presentation of results | | | | | | | | | | | | | | | | |
| a) Objectives and principles of documentation | | | | ■ | | | | | | | | | | | | |
| b) Organization of documentation | | | | | | ■ | | | | | | | | | | |
| c) Preparation of documentation | | | | | | ■ | ■ | ■ | ■ | ■ | ■ | ■ | ■ | ■ | ■ | ■ |
| Quality control and review | | | | | | | | | | | | | | | | |
| 16 Performance of internal and external peer reviews | | | | | | | ■ | ■ | ■ | ■ | ■ | ■ | | | | |
| 17 Integration of results of internal and external peer reviews | | | | | | | | | ■ | ■ | ■ | ■ | ■ | ■ | | |

## ANNEX III EXAMPLES OF COMMON RISK METRICS IN LEVEL 2 PROBABILISTIC SAFETY ASSESSMENT

III-1. Large release frequency and large early release frequency are the most common measures of risk used in Level 2 PSA. In many Member States, numerical values of this type are used as probabilistic safety goals or criteria. For example, Level 2 PSA risk metrics for large early release frequency should provide information with regard to the frequency of the release, on the release category with regard to the main radioactive material in that release category and the notion of the time of the release. large release frequency should be used as an integral indicator of the risk profile covering early and late radioactive releases. Level 2 PSA risk metrics large release frequency should provide information with regard to both the frequency of the release and on the release categories with regard to the main radioactive materials in that release categories integrated over a period of time.

The following tables provide examples of large release frequency and large early release frequency values and definitions in some Member States, with the reference from where such information comes from.

TABLE III–1. Examples of Member States practice on large release frequency risk metrics / safety goals.

| Member State | Reference | Large release frequency risk metrics Definition | Safety goal frequency, 1/r.y. |
|---|---|---|---|
| Canada | [III-1] | 100 TBq of Cs-137 | $< 1\cdot10^{-6}$ |
| | | 1000 TBq of I-131 (Small release frequency) | $< 1\cdot10^{-5}$ |
| Czech Republic | [III-2] | A severe accident, which could lead to an early radiation accident or a large radiation accident, is a practically eliminated matter | Radiation accidents are to have a very low frequency of occurrence, which means the occurrence of internal postulated initiating events or scenarios over a period 100 times longer than the lifetime of the nuclear installation; internal postulated initiating events and scenarios of design extension conditions are to be included in this category |
| Bulgaria | [III-4] | If for Cs-137 in 30km zone > 30TBq,<br>If evacuation ends after 12/24/48 hours | $\leq 10^{-5}$ |
| Finland | [III-5] | 100 TBq of Cs-137 | $< 5\cdot10^{-7}$ |
| France | [III-17] | Primarily for new nuclear power plant designs: Protective measures for the public should be very limited in terms of extension and duration, meaning no permanent relocation, no evacuation needed outside of the immediate vicinity of the plant site, neither sheltering nor long-term restriction of food consumption outside the vicinity of the plant site. Consequently, these accidents should not lead to neither contamination of large areas nor long-term environmental pollution. | [No quantitative value] |
| Japan | [III-8] | 100 TBq of Cs-137 | $< 1\cdot10^{-6}$ |
| Russian Federation | [III-9] | The release of radioactive substances into the environment during an accident at NPP, when in case of exceeding established criteria for radiation doses it is necessary to implement measures to protect the population within the initial stage of the accident (up to 10 days) on the border of the protective actions planning zone and outside it.<br>It should be noted that established frequency of release it is not a safety goal, it is safety target. | $< 1\cdot10^{-7}$ |
| Slovak Republic | [III-10] | > 1% of Cs-137 released from the core inventory | |
| Switzerland | [III-11] | >200 TBq of Cs-137 per calendar year | |

| Member State | Reference | Large release frequency risk metrics Definition | Safety goal frequency, 1/r.y. |
|---|---|---|---|
| Ukraine | [III-20] | Large release is defined as requiring public evacuation at the boundary of the protection area | criterion / goal for existing plants: < 1·10-6 1/r.y.; criterion / goal for new plants: < 1·10-7 1/r.y. |
| USA | [III-12] | Consistent with the traditional defence-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation. | $< 1 \cdot 10^{-6}$ |
| USA | [III-16] | Expected number of large releases per unit of time considering that large release is defined as the release of airborne fission products to the environment such that there are significant off site impacts. Large release and significant off site impacts may be defined in terms of quantities of fission products released to the environment, status of fission product barriers and scrubbing, or dose levels at specific distances from the release, depending on the specific analysis objectives and regulatory requirements. | |

TABLE III–2. Examples of Member States practice on LERF definition.

| Member State | Reference | LERF risk metrics definition | Safety goal frequency, 1/r.y. |
|---|---|---|---|
| Canada | [III-1] | (for operating NPP) Consistent with INSAG-12.<br>(for new NPP) The sum of frequencies of all event sequences that can lead to a release to the environment of more than $10^{14}$ becquerel of cesium-137 is less than $10^{-6}$ per reactor year. A greater release may require long term relocation of the local population. | $< 1{\cdot}10^{-5}$ for operated NPP<br>$< 1{\cdot}10^{-6}$ for new NPP |
| Czech Republic | [III-3] | More than >1% of Cs137 of the core inventory released to the environment within 10 hours after the beginning of the severe accident (T cladding = 1200°C) | $< 1{\cdot}10^{-6}$ for new NPP |
| Bulgaria | [III-4] | If for Cs-137 in 30km zone > 30TBq<br>if evacuation ends before 12/24/48 hours | $< 1{\cdot}10^{-5}$ for operated NPP<br>$< 1{\cdot}10^{-6}$ for new NPP |
| Finland | [III-5] | The accident sequences, in which the containment function fails or is lost in the early phase of a severe accident, have only a small contribution to the reactor core damage frequency.<br>Early means that there is no time to implement the warning and protective measures prior to the release. An exact number of hours has not been defined but warning and protection are typically estimated to take approximately four hours after the rescue department receives information on the need to take shelter. (Guide VAL 1.) The objective is that protective measures are not needed in a situation in which there would practically be no time to implement them. | |
| France | [III-17] | The objective of the design is that event sequences that could lead to large releases with a kinetics that might not allow the timely implementation of the measures necessary for the protection of populations should be physically impossible to happen and if not, very unlikely to happen with a high degree of confidence. | |
| Hungary | [III-6] | Radioactive release in the case of which urgent precautionary measures are required off the site but no sufficient time is available for their introduction | $< 1{\cdot}10^{-5}$ for operated NPP ($10^{-6}$ target)<br>$< 1{\cdot}10^{-6}$ for new NPP |
| Korea, Republic of | [III-7] | The frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. | $< 1{\cdot}10^{-5}$ for operated NPP<br>$< 1{\cdot}10^{-6}$ for new NPP |
| Pakistan | [III-13] | | $< 1{\cdot}10^{-6}$ |

| Member State | Reference | LERF risk metrics definition | Safety goal frequency, 1/r.y. |
|---|---|---|---|
| Russian Federation | [III-14] | Decisions on measures to protect the population in the event of a major radiation accident with radioactive contamination of the territory are made on the basis of a comparison of the predicted dose prevented by the protective measure and the levels of contamination over a period of 10 days | Term LERF is not defined in the Russian Federation. |
| Slovak Republic | [III-18] | More than > 1% of Cs137 released from the core inventory to the environment within 10 hours after the beginning of the IE | $< 1\cdot10^{-5}$ for operated NPP<br>$< 1\cdot10^{-6}$ for new NPP |
| Switzerland | [III-19] | LERF is expected number of events per calendar year with a release of more than $2\cdot10^{15}$ Bq of Iodine-131 per calendar year within the first 10 hours after core damage. | $< 1\cdot10^{-5}$ for operated NPP<br>$< 1\cdot10^{-6}$ for new NPP |
| USA | [III-15] | LERF is defined as the sum of the frequencies of those accidents leading to rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is the potential for early health effects. (Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.) | $< 1\cdot10^{-5}$ |
| USA | [III-16] | Expected number of large early releases per unit of time, where a large early release is defined as a large release occurring before the effective implementation of off-site emergency response and protective actions and there is the potential for early health effects. | |

Additional information on risk metrics used in other countries could be found in [III-21].

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