IAEA
International Atomic Energy Agency

IAEA SAFETY STANDARDS

No. SSG-4 (Rev. 1)

for protecting people and the environment

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

SPECIFIC SAFETY GUIDE

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available on the IAEA web site:

http://www-ns.iaea.org/standards/

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

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The **IAEA Nuclear Energy Series** comprises informational publications to encourage and assist research on, and the development and practical application of, nuclear energy for peaceful purposes. It includes reports and guides on the status of and advances in technology, and on experience, good practices and practical examples in the areas of nuclear power, the nuclear fuel cycle, radioactive waste management and decommissioning.

DEVELOPMENT AND APPLICATION OF LEVEL 2 PROBABILISTIC SAFETY ASSESSMENT FOR NUCLEAR POWER PLANTS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

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DEVELOPMENT AND APPLICATION OF LEVEL 2 PROBABILISTIC SAFETY ASSESSMENT FOR NUCLEAR POWER PLANTS

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2025

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Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
tel.: +43 1 2600 22529 or 22530
email: sales.publications@iaea.org
www.iaea.org/publications

Printed by the IAEA in Austria
June 2025
STI/PUB/2105
https://doi.org/10.61092/iaea.buld-d6ia

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Development and application of level 2 probabilistic safety assessment for nuclear power plants / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2025. | Series: safety standards series, ISSN 1020–525X ; no. No. SSG-4 (Rev. 1) | Includes bibliographical references.
Identifiers: IAEAL 25-01739 | ISBN 978-92-0-137424-0 (paperback : alk. paper) | ISBN 978-92-0-137524-7 (pdf) | ISBN 978-92-0-137624-4 (epub)
Subjects: LCSH: Nuclear power plants — Safety measures. | Nuclear power plants. | Nuclear reactors — Safety measures. | Nuclear power plants — Risk assessment.
Classification: UDC 621.039.58 | STI/PUB/2105

FOREWORD

by Rafael Mariano Grossi
Director General

The IAEA's Statute authorizes it to "establish…standards of safety for protection of health and minimization of danger to life and property". These are standards that the IAEA must apply to its own operations, and that States can apply through their national regulations.

The IAEA started its safety standards programme in 1958 and there have been many developments since. As Director General, I am committed to ensuring that the IAEA maintains and improves upon this integrated, comprehensive and consistent set of up to date, user friendly and fit for purpose safety standards of high quality. Their proper application in the use of nuclear science and technology should offer a high level of protection for people and the environment across the world and provide the confidence necessary to allow for the ongoing use of nuclear technology for the benefit of all.

Safety is a national responsibility underpinned by a number of international conventions. The IAEA safety standards form a basis for these legal instruments and serve as a global reference to help parties meet their obligations. While safety standards are not legally binding on Member States, they are widely applied. They have become an indispensable reference point and a common denominator for the vast majority of Member States that have adopted these standards for use in national regulations to enhance safety in nuclear power generation, research reactors and fuel cycle facilities as well as in nuclear applications in medicine, industry, agriculture and research.

The IAEA safety standards are based on the practical experience of its Member States and produced through international consensus. The involvement of the members of the Safety Standards Committees, the Nuclear Security Guidance Committee and the Commission on Safety Standards is particularly important, and I am grateful to all those who contribute their knowledge and expertise to this endeavour.

The IAEA also uses these safety standards when it assists Member States through its review missions and advisory services. This helps Member States in the application of the standards and enables valuable experience and insight to be shared. Feedback from these missions and services, and lessons identified from events and experience in the use and application of the safety standards, are taken into account during their periodic revision.

I believe the IAEA safety standards and their application make an invaluable contribution to ensuring a high level of safety in the use of nuclear technology. I encourage all Member States to promote and apply these standards, and to work with the IAEA to uphold their quality now and in the future.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered 'overarching' requirements, are expressed as 'shall' statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it

[1] See also publications issued in the IAEA Nuclear Security Series.

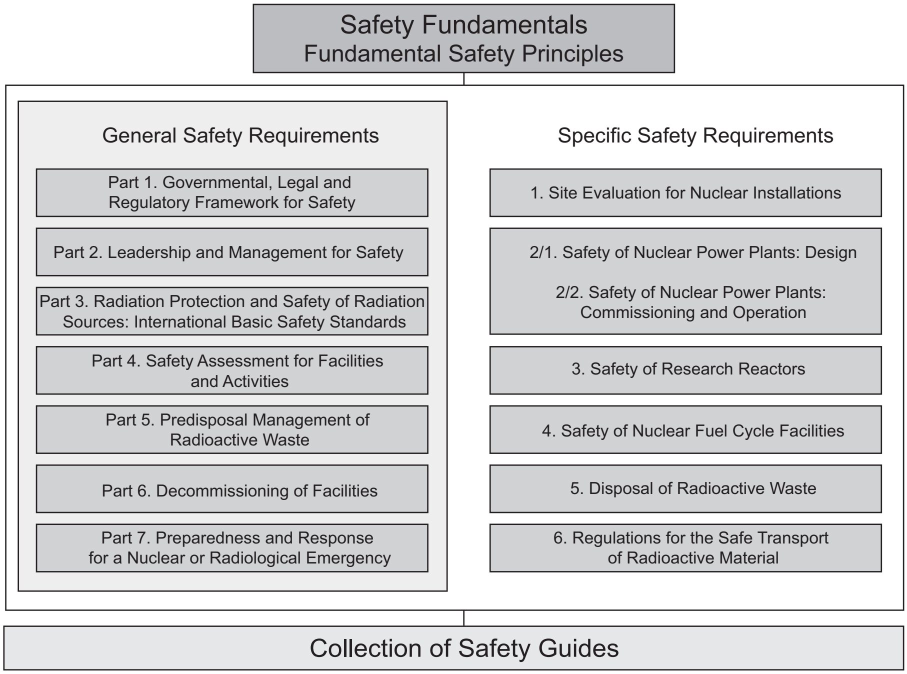

FIG. 1. The long term structure of the IAEA Safety Standards Series.

is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five Safety Standards Committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the Safety Standards Committees and may provide comments on draft standards. The membership of the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards.

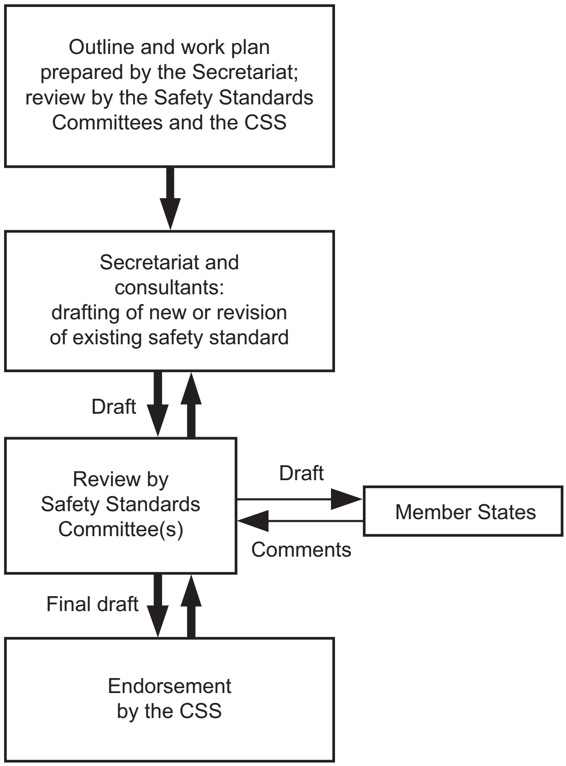

FIG. 2. The process for developing a new safety standard or revising an existing standard.

It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as they appear in the IAEA Nuclear Safety and Security Glossary (see https://www.iaea.org/resources/publications/iaea-nuclear-safety-and-security-glossary). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

CONTENTS

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 control the inherent risk posed by nuclear installations. In particular, para. 3.22 of SF-1 [1] 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 Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2], states that "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. Requirement 15 of GSR Part 4 (Rev. 1) [2] states that "**Both deterministic and probabilistic approaches shall be included in the safety analysis.**" 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 probabilistic safety assessment (PSA) contributes to assessing and verifying that a balanced design of the nuclear power plant has been achieved in relation to the overall risk from potential internal initiating events and internal and external hazards, and to preventing cliff edge effects.

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 of the procedures in place or envisaged to prevent core and/or fuel damage. Further recommendations are provided in IAEA Safety Standards Series No. SSG-3 (Rev. 1), 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 radioactive material to the environment. This analysis provides additional insights into the relative importance of accident prevention and mitigation measures and the physical barriers (e.g. containment) to the release of radionuclides to the environment.

(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. Each PSA level could also have a different scope depending on the range of initiating events (internal and/or external to the plant) and modes of operation of the plant that are to be considered.

1.8. Level 2 PSA is a structured process. Although there may be differences in the approaches to performing a Level 2 PSA (see para. 2.6), the general main steps, in the context of the overall PSA (see para. 1.6), are shown in the two central boxes of Fig. 1 and can be described as follows:

(a) Level 1 PSA provides information on the accident sequences that lead to core and/or fuel damage and hence provides the starting point for Level 2 PSA. The accident sequences identified by Level 1 PSA might not include information on the status of the SSCs dedicated to ensuring the confinement function (e.g. the containment associated 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 core and/or fuel damage are grouped into plant damage states (PDSs) based on similarities in the plant conditions that determine the further accident progression. Some extended event trees can complete the information provided by Level 1 PSA.

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

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[1] Accident progression event trees are also termed 'containment event trees'. The term 'accident progression event tree' has been chosen throughout this Safety Guide, as was done in the ASAMPSA2 project [5], because it is more generally applicable.

(d) Source term[2] analysis is used to determine the quantities and timings of radioactive material released to the environment from each release category[3].

1.9. For practical purposes, a number of grouping tasks sometimes need to be performed for Level 2 PSA. Therefore, the process for carrying out Level 2 PSA depends on the approach selected for the grouping indicated in the two central boxes of Fig. 1, as described below:

(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 use a multi-step process of grouping and regrouping 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.

1.10. Further grouping or regrouping of the release categories into a condensed set[4] to be used for Level 3 PSA may be needed. The interface between Level 2 PSA and Level 3 PSA is not addressed in detail in this publication, although it is briefly mentioned in Section 15. Level 1 PSA and Level 2 PSA of varying scopes and levels of detail have been performed for almost all nuclear power

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[2] The term 'source term' is to be understood as defined in the IAEA Nuclear Safety and Security Glossary [6] as "The amount and isotopic composition of radioactive material released (or postulated to be released) from a facility." This term is 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. A more detailed definition provided in Ref. [7] is as follows:

> "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.)".

[3] A release category is a group of accident progression sequences that would generate a similar source term for release to the environment. The categories are defined by attributes in relation to the release (see Section 10).

[4] 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 the term 'source term category' is generally used synonymously with 'release category' and is understood by the definition provided in footnote 3. See also footnote 15.

plants in operation or under construction worldwide, whereas Level 3 PSA has been performed only for some nuclear power plants in some States.

1.11. This Safety Guide was prepared on the basis of a systematic review of relevant IAEA publications, including Refs [1–4, 8, 9] and an International Nuclear Safety Advisory Group report [10].

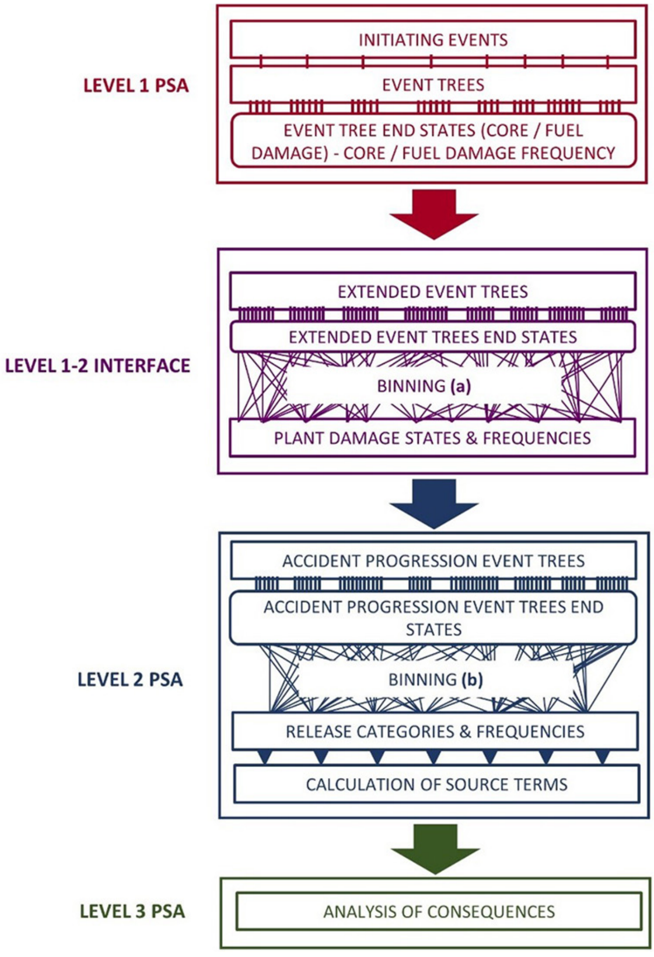

FIG. 1. General overview of the development of a typical Level 2 PSA in the context of the overall PSA.

1.12. This Safety Guide supersedes IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants[5].

OBJECTIVE

1.13. The main objective of this Safety Guide is to provide recommendations for meeting the requirements of GSR Part 4 (Rev. 1) [2] on performing or managing a Level 2 PSA project and the development of the Level 2 PSA for a nuclear power plant; this Safety Guide therefore complements SSG-3 (Rev. 1) [4]. It aims to promote a standard framework, standard terms and a standard set of documents for PSAs to facilitate regulatory and external peer review, in particular for Level 2 PSA results. Another objective is to provide a consistent, reliable means of ensuring the effective fulfilment of obligations under Article 14 of the Convention on Nuclear Safety [11].

1.14. 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 in order 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 the phenomenology for other particular reactor technologies needs to be investigated and identified. For example, for molten salt reactors with liquid fuel, the concept of core melt is not meaningful.

SCOPE

1.15. 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

[5] 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).

elements of the PSA rather than on details of the modelling methods. This Safety Guide includes all of 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.

1.16. 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) events. If the objectives and scope of the Level 2 PSA are limited, only the relevant recommendations provided in this Safety Guide apply.

1.17. If the aim of the PSA is to determine all of the contributions to risk to public health and society, in the calculation of the source term the PSA needs to take into account the potential for release from other sources of radioactivity at the plant, such as irradiated fuel and stored radioactive waste. Such an aim is not covered by this Safety Guide, which focuses on releases of radioactive material resulting from severe accidents in the reactor and the spent fuel pool (SFP). This Safety Guide also covers the development of a multi-unit Level 2 PSA for sites where several units are located and is suitable for use where regulatory requirements compel such an assessment as part of the quantification of the source term at the site level.

1.18. 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 phenomena associated with the nuclear reactor technology used.

1.19. Recommendations relating to the performance, project management, documentation and peer review of a PSA and the implementation of a management

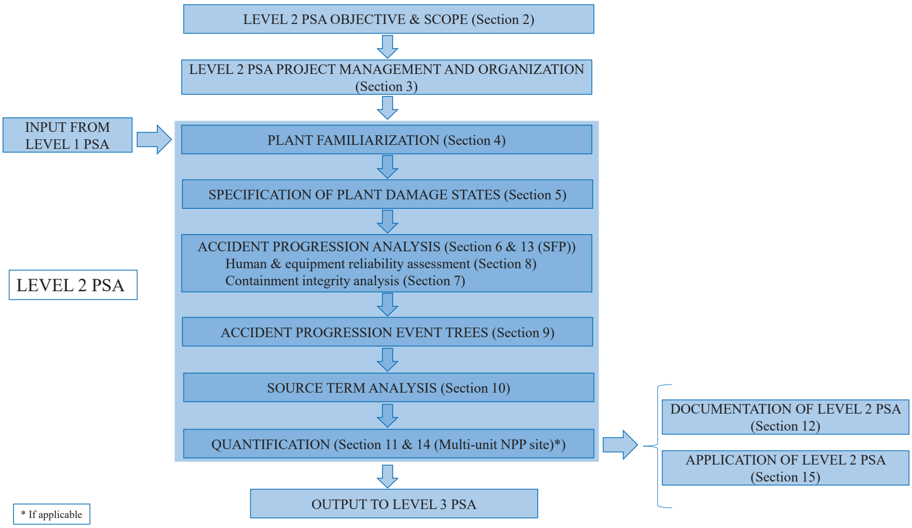

FIG. 2. Main steps in the performance of Level 2 PSA.

system in accordance with IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [8] are provided in SSG-3 (Rev. 1) [4] and are therefore not addressed in detail here. This Safety Guide addresses only the aspects of PSA that are specific to Level 2 PSA.

STRUCTURE

1.20. 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 SFP. Section 14 provides recommendations on the performance of Level 2 PSA for a site with multiple nuclear power plants. Section 15 provides recommendations on the use and applications of a Level 2 PSA. The Appendix gives an overview of human reliability analysis in Level 2 PSA. Annex I describes various types of computer code available for simulation of severe accidents and PSA studies. Annex II presents a sample plan of activities and an 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.

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 whether there are sufficient safety provisions[6] that have been designed in a balanced way to manage a severe accident and to mitigate the effects of such an accident to ensure that sufficient protection of people and the environment has been achieved. Level 2 PSA, in combination with Level 1 PSA, contributes to demonstrating the practical elimination[7] of plant event sequences that could lead to an early radioactive release or a large radioactive release. Recommendations on the implementation of selected requirements in SSR-2/1 (Rev. 1) [3] related to defence in depth and practical elimination of plant event sequences leading to early radioactive releases or large radioactive releases are provided in SSG-88 [12]. The sufficiency of these provisions is normally demonstrated by compliance with numerical safety goals (see paras 2.16 and 2.17), whereas the balanced design is demonstrated by analysis of the individual contributions of the safety provisions to the overall risk

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[6] For the purposes of this Safety Guide, ‘safety provisions’ are considered to be design solutions applied to structures, systems and components and related operational strategies.

[7] The concept of practical elimination applies to plant event sequences that could lead to unacceptable consequences (i.e. an early radioactive release or a large radioactive release) that cannot be mitigated by reasonably practicable means. Practical elimination implies that those plant event sequences have to be demonstrated to be either physically impossible or, with a high level of confidence, extremely unlikely to arise by the implementation of safety provisions in the form of design and operational features (see IAEA Safety Standards Series No. SSG-88, Design Extension Conditions and the Concept of Practical Elimination Concept in the Design of Nuclear Power Plants [12]).

profile. Safety provisions to manage severe accidents and mitigate their effects could include the following:

(a) Systems provided specifically to mitigate the effects of a 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 of non-permanent equipment (e.g. diesel power generators, pumps) for accident management;
(c) Guidance to plant operators on severe accident management.

2.3. The objectives of Level 2 PSA should be defined. These might 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 and into the decision making process for a given application.
(d) To provide an input into determining compliance with probabilistic safety goals, or with probabilistic safety criteria, if these have been set. The 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.
(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 radioactive 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 for the plant.
(l) 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.
(n) To inform the choice of representative severe accidents for deterministic analysis.

2.4. 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 due to the difficulty or impossibility of implementing design safety features to manage severe accidents at a later stage.

SCOPE AND APPROACHES FOR 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.**"

Furthermore, 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 in which the Level 1 and Level 2 PSA models are developed, linked and quantified in a single software tool. The second is a separated approach, in which 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 and results from Level 1 to Level 2 are needed. Reference [5] provides information on the advantages and disadvantages of each approach. 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 fully takes 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 a Level 2 PSA should be determined by its defined objectives (see paras 2.2 and 2.3) and its specific intended uses and applications (see para. 15.2). Appropriate consideration should be given to the significance of key uncertainties associated with phenomena and modelling (see Section 11). 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 internal events. Such a base model should be used as a basis for extension to internal and external hazards.

2.9. If the starting point of a Level 2 PSA is an existing Level 1 PSA, then its output might not explicitly cover all of 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 (e.g. the containment and its associated systems) might 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. [13])), 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, and damage of containment structures due to seismic events or transportation accidents.

2.11. If the SFP is located inside the same containment as the reactor (e.g. in some pressurized water reactor and boiling water reactor designs), the Level 2 PSA should consider the combined consequences of severe accident phenomena induced by the reactor core and the SFP for the containment and source term calculations. Recommendations on Level 2 PSA for the SFP are provided in Section 13.

2.12. If the scope of the PSA includes sources of radioactivity other than the reactor and the SFP (e.g. refuelling pool, transport casks, liquid radioactive waste, dry long term storage facility for spent fuel) that are located outside of the containment (e.g. reactor containment), then the potential risk of release from those sources should be considered. As stated in para. 1.17, releases from other sources of radioactivity at the plant, such as irradiated fuel and stored radioactive waste, are not considered 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 should include an assessment of uncertainties, consistent with the intent and scope of the study being undertaken.

2.14. If the scope of the PSA study considers Level 3 PSA, the scope of the Level 2 PSA should consider the inputs 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. Recommendations on conducting a Level 2 PSA for multi-unit nuclear power plants are provided in Section 14.

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 provided 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 of this Safety Guide 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 above a certain 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 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. In defining Level 2 PSA risk metrics, adequate information should be provided to understand the meaning of qualitative concepts or terms to be derived as quantitative values. In particular, the terms ‘large’, ‘early’, ‘release’, ‘exceedance’ and ‘frequency’ should be taken into account. When defining Level 2 PSA risk metrics, the following should be considered:

(a) The defined limits related to unacceptable consequences in terms of dose to the general public and the impact on the environment regarding limits in space and time;
(b) The evaluation of capabilities associated with severe accident management programmes and with the emergency 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 radionuclides contained in the fuel, the reactor core and the reactor coolant system;
(d) The radionuclides (and their radiotoxicity) that could be liberated as a result of the accident sequences;
(e) The expected probability of occurrence of severe accidents within a specific time frame;
(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) Probabilistic safety goals or criteria currently in use in other States;
(b) Operating experience feedback;
(c) The relationship between defined safety goals and different PSA levels (e.g. between core damage frequency and large early release frequency);

(d) Implications of the 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) [14] 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 design of SSCs 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 accidents 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 periodic updating is termed a 'living PSA'. In updating a PSA, account should be taken of changes in the design and operation of the plant (e.g. operating procedures and practices, emergency operating procedures, maintenance policies, operator training and accident management practices), changes to external facilities or sources of external hazards, new technical information, more sophisticated PSA methods and tools that become available, changes in industry operating experience and new plant specific data derived from the operation of the plant (e.g. data to be used for the assessment of initiating event frequencies or component failure probabilities). The updating of a PSA should be initiated by a specified process, and the status of the PSA should be reviewed periodically 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 and, if necessary, 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 as an aid to 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 this type of living PSA can be of substantial benefit to the operating organization, and its use is generally welcomed by regulatory bodies [15].

USE OF PSA IN THE DECISION MAKING PROCESS

2.24. Although the scope of the Safety Guide is limited to consideration of Level 2 PSA, this section describes some issues relevant to PSA in general, in order to provide a complete picture of the capabilities of the 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 2.26–2.27 provide recommendations on meeting Requirement 1 of GSR Part 4 (Rev. 1) [2], which relates to a graded approach to the scope and level of detail of safety assessment, and Requirement 14 of GSR Part 4 (Rev. 1) [2], which relates to the scope of the safety analysis for a PSA.

2.26. Quantitative results of PSA are often used to verify compliance with probabilistic safety goals or criteria, which, depending on the scope of the PSA, are usually formulated in terms of quantitative estimates of core damage frequency or fuel damage frequency, frequencies of different types of radioactive release 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 (i.e. their scope and level of detail). Therefore, in order to use the PSA results for 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 operational states that are not in the model does not threaten compliance with the probabilistic safety goals or criteria.

2.27. 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.

2.28. 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.**"

2.29. 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.

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

(a) Design modifications and plant modifications;
(b) Optimization of plant operation and maintenance;
(c) Safety analysis and research programmes;
(d) Regulatory issues;
(e) Scenarios to be focused on during emergency preparedness drills;
(f) Development of severe accident management guidelines;
(g) Optimization of training.

2.31. 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. 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 [16]. If numerical results from the PSA are to be used, reference values against which these results can be compared should be established.

2.32. 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. These reviews 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 2.20–2.23.

2.33. 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. Such decisions 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.

2.34. 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. Such a comparison should be performed for all probabilistic safety goals or criteria defined for the plant, including those that address system reliability, core damage frequency, fuel damage frequency, radioactive release frequency, health effects for the public and other off-site consequences such as land contamination and restrictions on foodstuffs. If numerical probabilistic safety criteria or goals are not specified, risk reduction possibilities can nevertheless be examined based on the Level 2 PSA results.

2.35. The PSA should aim to identify all accident sequences that contribute to risk in a non-negligible way. If the PSA does not address all significant contributions to risk, or if its scope is reduced (see para. 2.25) (e.g. if it omits external hazards or shutdown states), then the insights from the PSA about the level of risk from the plant, the balance of the risk contributors and the need for changes to be made to the design or operation to reduce risk might be limited. Such limitations should be acknowledged when using PSA to support decision making and should be addressed by alternative analyses as necessary.

2.36. The results of the PSA should be used to identify weaknesses in the design or operation of the plant and in actions considered in severe accident management strategies (see also IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [17]). These weaknesses can be identified by considering the contributions to the risk from groups of initiating events and the importance measures[8] for SSCs and for human errors. If 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. PROJECT MANAGEMENT AND ORGANIZATION FOR PSA

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 for PSA provided in section 3 of 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 OBJECTIVES OF THE LEVEL 2 PSA PROJECT

3.2. Paragraphs 3.3–3.5 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.

3.3. Differing end uses place differing emphases and needs 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 should be in agreement with the main objective of the Level 2 PSA and intended purposes, as described in Section 1.

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[8] Typical importance measures used in probabilistic safety assessment are Fussell–Vesely importance, risk reduction worth, risk achievement worth and Birnbaum importance (see para. 5.171 of SSG-3 (Rev. 1) [4]). These importance measures provide 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.

3.4. 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 the objectives, intended uses and applications of the Level 2 PSA all being taken into account.

3.5. The objectives of the Level 2 PSA project should be understandable and achievable by the users of the Level 2 PSA. In this context, previous experiences from the management of other Level 2 PSA projects are very valuable and should be gathered.

DETERMINATION OF THE SCOPE OF THE LEVEL 2 PSA PROJECT

3.6. Paragraphs 3.7–3.8 provide recommendations on meeting Requirements 1 and 14 of GSR Part 4 (Rev. 1) [2] in relation to the scope of the Level 2 PSA project.

3.7. The scope of the Level 2 PSA project should be determined by the overall scope of 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 defining the scope and the methods used for modelling the severe accident phenomena and for the contribution of the SSCs to the risk of a radioactive release depending on their source (see para. 1.17). A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs that are part of an installation containing other potential sources of radioactive releases (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).

3.8. In compliance with para. 2.14, consideration should be given to the input requirements for a Level 3 PSA, as applicable, when determining the scope of the Level 2 PSA project. The ultimate product of a Level 2 PSA is 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 includes 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 are a description of a number of challenges to the dedicated SSCs ensuring the confinement function (e.g. the containment), and a description of the possible responses of those SSCs.

PARTICULAR FEATURES OF PROJECT MANAGEMENT FOR LEVEL 2 PSA

3.9. 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 3.3–3.9 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 reflects realistic operating practices to the extent possible and that it does take account of recent developments in methods, models and data.

3.10. 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.

3.11. 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 many resources are employed, analyses of both the behaviour of the containment during the severe accident and the radiological source terms are subject to large uncertainties associated with phenomena. The expertise and technical resources needed should be considered, in accordance with the scope of the Level 2 PSA project, in the selection of computer codes (see paras 3.17 and 3.18), and in the selection, training and qualification of personnel (see paras 3.19–3.22).

3.12. To meet the requirements established in GSR Part 2 [8], 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 team for establishing independent review processes or performing comparative studies, as appropriate (see paras 3.24–3.29). Recommendations on the technical review of relevant aspects of the analysis, project documentation and configuration control are provided in Section 12.

3.13. The Level 2 PSA project management team should aim to ensure that the insights gained from performing 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.

3.14. 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 team are defined in paras 3.15 and 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 paras 3.24–3.29).

3.15. Taking into account the stage in the plant lifetime (e.g. design, operation) and the objectives to be reached, the Level 2 PSA project management team should specify what prior information is key for the successful development of the Level 2 PSA. Such information might be provided, for example, on the following:

(a) Selection of staff and responsibilities (see paras 3.19–3.22);
(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 3.6–3.8);
(c) Planning and scheduling of project activities, including identifying the need for research related activities, software development, verification and validation, and training;
(d) Availability and collection of plant data in relation to SSCs, severe accident phenomena, human factors, emergency operating procedures and/or severe accident management guidelines, and internal and external hazards [13];
(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;
(h) Scope and structure of the documentation for the Level 2 PSA (see Section 12).

SELECTION OF SOFTWARE, APPROACHES AND METHODS

3.16. 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 for PSA in general provided in paras 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 advantageous to have the possibility to deal with multiple point branches as well as with correct quantification of success branches.

3.17. The computer codes selected for use 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 progression analysis for Level 2 PSA should be verified and validated by experiments and/or benchmarking to ensure a sufficient level of confidence in 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 progression analysis codes is much lower than for other codes used to support PSA, such as the thermohydraulic codes used to support the success criteria for the credited systems[9] 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 conduct 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.

3.18. The selection of an integrated or separated approach (see para. 2.6) and the methods for accident progression event tree construction (see Section 9) should be consistent with the scope and objectives of the Level 2 PSA project.

TEAM SELECTION FOR A LEVEL 2 PSA PROJECT

3.19. 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:

(a) Knowledge of the design and operation of the plant;
(b) Knowledge of severe accident phenomena and challenges to the containment;

[9] ‘Credited systems’ are systems credited in the PSA, which include operating and standby safety systems and non-safety systems whose operation during an accident can help prevent an undesired end state (e.g. core damage, fuel damage). Also ‘credited SSCs’ is a term used in this publication to specify particular structures or components credited in the PSA.

(c) Knowledge of physics regarding radioactive material;
(d) 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 performed, the scope of the Level 2 PSA and the intended applications of the Level 2 PSA. To the extent possible, there should be extensive participation of the plant engineers and utility personnel or designers (e.g. if performed at the design stage) as well as probabilistic safety analysts specialized in severe accident phenomena and other Level 2 PSA disciplines.

3.20. The Level 2 PSA project management team should provide working arrangements that ensure good interaction and communication between all of the members of the team, including project managers and analysts. Effective communication throughout the project is essential. In addition, the Level 2 PSA project management team 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 balance of efforts across all topics.

3.21. The following recommendations apply to Level 2 PSA team members using computer codes:

(a) They should have adequate training on the computer code(s) used.
(b) They should have a sound knowledge of the models and methods implemented in the computer code(s).
(c) They should have sufficient understanding of the limitations of the computer code(s) in relation to the phenomena to be modelled.
(d) They should be familiar with the guidance on the computer code(s) and the procedures for implementing the models in the computer codes.
(e) They should have adequate capacity to evaluate the results of the computer code(s).

3.22. For a nuclear power plant in operation, the following members should be considered for inclusion in the Level 2 PSA team:

(a) A technical project leader responsible for coordination among all of the project experts;
(b) Experts in, and staff working on, the design and operation of the plant (particularly the containment associated 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 load bearing 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, 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

3.23. 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.**"

3.24. The main objective of the independent verification of the Level 2 PSA is to confirm that the methods and approaches selected, the 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 into project management when developing the Level 2 PSA. This internal independent verification, if conducted, should look at each aspect of the Level 2 PSA (e.g. data and computers codes used, interface with Level 1 PSA, supporting studies, assumptions in the event trees, event tree quantifications, results interpretation) to ensure that it is of a sufficient quality for its purposes. This internal independent verification process may help to identify some sources of uncertainty (see, e.g., paras 5.14, 6.24–6.27, 7.23–7.29, 8.19–8.23. and 11.21–11.24).

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

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

3.27. Level 2 PSA uses 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 should ensure that the group of experts or institution in charge of the independent verification has an adequate level of expertise to satisfactorily evaluate the data, assumptions and models (i.e. deterministic and probabilistic) considered in the Level 2 PSA (see para. 3.29).

3.28. The results of the independent verification of the Level 2 PSA should be reported in a separate document, which is to be presented to the regulatory body upon request.

3.29. The report compiling the results of the independent verification of the Level 2 PSA should consider the assessment of the technical adequacy of the following:

(a) The development, grouping and quantification of PDSs;
(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 development, grouping, quantification and source term characterization of the release categories;
(g) Supporting calculations, correct and appropriate application of codes;
(h) Structural analysis and/or 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 the sensitivity analysis performed (e.g. the bases for the selection of probability distributions of uncertain parameters and assignment of their distribution 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 entire 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 depend on the particular reactor design and technology, and might include coolers employing fans or water, the containment heat removal system, containment sprays and/or filtered containment venting systems, the containment exhaust system and suppression pools, a dedicated set of steam relief valves and hydrogen control systems (i.e. ignitors, recombiners). This familiarization process should cover the reactor building, auxiliary building, secondary containment and other relevant structures and buildings, which differ depending on the reactor technology and design. For existing plants, familiarization with the plant should include a plant walkdown with the participation of operating personnel and plant technical staff. Interviews with operating personnel and plant technical staff fulfilling relevant roles from a Level 2 PSA perspective should also be conducted.

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. This 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 spreads 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 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. This limits 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

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

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	Set point and procedures
Pressure relief capacity	Actual operational values
Isolation of containment penetrations connected to the reactor coolant system	Potential for containment bypass
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

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

Key plant and/or containment design feature	Comment
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, metallic liner and other
Operating pressure and temperature	Actual operational values
Hydrogen control mechanisms	Provision of inertness, ignitors, passive recombiners, other
Suppression pool volume	Water available for containment pressure control or fission product retention
Containment cooler capacity and set points	Actual operational values
Containment associated systems	Actual operational values, logic
Concrete aggregate of each containment structure	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 (e.g. for painting or pipe insulation) potentially affecting sump filter clogging
Heat removal paths from reactor and containment	Layout, location and operation
Configuration of heat sink	Operational procedure

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

Key plant and/or containment design feature	Comment
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, 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	The design of some plants (recent or backfitted) includes 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, rack design, loading pattern (if any)
Decay heat	Total decay heat in normal storage conditions and for emergency unloaded core
Radioactive material inventory	Full inventory of radionuclides in the fuel stored in the SFP
Design parameters	Nominal coolant temperature and level, maximum allowed coolant temperature, minimum allowed coolant level

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

Key plant and/or containment design feature	Comment
Safety features	Nominal and minimum flow rate, coolant inventory, soluble absorber concentration, nominal and maximum temperature of the coolant
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 needs 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 INSTALLATIONS WITH RADIOACTIVE SOURCES ON A SITE

4.4. Paragraphs 4.5–4.9 are intended to provide 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 multiple installations with radioactive sources are located. Recommendations on Level 2 PSA for multi-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 multi-unit sites.

4.6. Performance shaping factors considered for the human reliability analysis should take into account 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 or external hazards and their combinations (see also Ref. [13]).

4.7. Considerations related to equipment (either installed or non-permanent) and systems that are shared among or common to all units on the site, 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 multi-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 4.11–4.14 provide an overview of the 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. As part of plant familiarization, the analyst should collect available documentation on the strategies implemented at the plant to cope with severe accident associated phenomena and should become familiar with the priorities and actions contained within these strategies. Strategies developed to cope with severe accident progression generally include those aimed at (a) protecting the confinement function, including preventing the containment bypass, and (b) if applicable, protecting the reactor building where the SFP is located. Depending on the reactor design, strategies should also address protection of the proper functioning of filtered venting systems in the auxiliary building, and management of leakage of liquid effluent from the reactor containment in case of recirculation of contaminated water outside the containment. During the progression of a severe accident involving the degradation of the fuel in the reactor vessel (e.g. in the reactor core for water cooled reactors), two important strategies are considered: first, 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 second, ex-vessel cooling and retention of damaged fuel (e.g. ex-vessel corium[10] cooling for some water cooled reactor designs) (see also paras 4.13 and 4.14).

4.12. The analyst should also be familiar with other strategies related to severe accident management. These strategies may include the SFP 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

[10] Corium is the material formed during the meltdown of a nuclear reactor. It is composed of nuclear fuel (uranium or plutonium) and material that melts on contact with the fuel.

and the control of radioactive releases from the containment. SSG-54 [17] 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 the availability of electricity, compressed air or water sources.

In-vessel melt retention

4.13. For water cooled reactors, the in-vessel melt retention strategy is intended to ensure a passive and/or active flooding of the reactor pressure vessel cavity up to a level high enough to ensure and maintain, with sufficient confidence, the integrity of the reactor pressure vessel by cooling it from outside and, if necessary, by additional cooling of the corium inside by in-vessel water injection. For other reactor 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). Information related to the success of this strategy should be collected during the familiarization task, for example on the following:

(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, this allows consideration of sufficient water circulation between the reactor pressure vessel wall and the insulation, and the evacuation path of the produced steam to the upper volume of the containment);
(d) Design solutions related to reactor pressure vessel internals (e.g. a large mass of steel in the corium relocated in the 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 time delay of corium arrival in the lower plenum and therefore reducing the residual heat removal).

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[11] The ‘focusing effect’ phenomenon involves a thin metal layer on top of a large oxidic pool. If the radiative heat transfer on top is inadequate to discharge the thermal power received from below, the temperature of the metal layer rises, and increasing amounts of energy flow are directed to the vessel wall. This focusing increases as the metal layer thickness decreases [18]. This effect can induce reactor vessel rupture.

Ex-vessel corium cooling

4.14. 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. Information related to the success of this strategy should be collected during the familiarization process, for example on the following:

(a) Analysed reactor vessel failure modes that support the strategy;
(b) Operator actions that support the strategy;
(c) The configuration of the corium spreading surface (relevant to reducing 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, 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 PROGRESSION ANALYSIS

4.15. 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 understanding of the plant design, phenomena[12] and features that might influence severe accidents and releases of radioactive material, the quantitative data that are necessary to perform the plant specific analysis should be collected and organized. The data necessary for the Level 2 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 necessary for the scope of the containment performance analysis.

[12] Phenomena could be obtained from a Phenomena Identification and Ranking Table analysis for severe accidents, if available.

4.16. Data should be obtained from sources such as the following:

(a) Design documents and/or plant licensing documents, such as the safety analysis report, technical specifications and system descriptions;
(b) As-built drawings;
(c) Plant specific normal operating, maintenance or test procedures;
(d) Information on automatic actuations in the plant;
(e) Emergency operating procedures and severe accident management guidelines;
(f) Engineering calculations or analysis reports;
(g) Observations during plant walkdowns 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.

4.17. 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 would therefore 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, considering that there are always differences from the reference plant, great care should 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
Ratio of zirconium mass to containment free volume	Potential for combustion and scaling of containment loads

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

Parameter and design feature	Significance or comparability
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 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 needs to be performed to provide the necessary input for the Level 2 PSA. The Level 1 PSA and Level 2 PSA interface depends on the 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. The PDSs should represent groups of accident sequences that have similar accident timelines, containment status and containment system (un)availability status and that 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 a radioactive release to the environment should be identified. The attributes of the PDSs provide boundary conditions for the performance of severe accident progression analysis.

5.3. The grouping of Level 1 PSA sequences into PDSs may involve some assumptions and simplifications that introduce additional uncertainties. The analyst should keep track of these assumptions and simplifications (see para. 11.22).

5.4. Care should be taken to ensure that assumptions that could minimize the source term 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 of the PDSs and validating their sum against the core and/or fuel damage frequency determined from Level 1 PSA. Justifications for any numerical deviations should be given.

5.5. The success criteria for a system credited in Level 2 PSA should specify the system mission time to reach a controlled state[13] or to fulfil the modelled system function. 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.

5.6. Paragraphs 5.7–5.25 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.

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

5.7. If the Level 2 PSA is developed following a separated approach (see para. 2.6), the Level 1 PSA does not account for specific aspects relevant to the specification of PDSs. For example, the Level 1 PSA might not have addressed the status of containment associated 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

[13] As defined in SSR-2/1 (Rev. 1) [3], a 'controlled state' is a plant state, following an anticipated operational occurrence or accident conditions, in which the fundamental safety functions can be ensured and which can be maintained for a time sufficient to effect provisions to reach a safe state.

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

Initiating event	Large loss of coolant accident Small loss of coolant accident Safety or relief valve stuck open Transient, such as: — Reactor trip — Loss of off-site power — Loss of electrical bus — Loss of feedwater — Loss of service water — Steam line break — Feedwater line break — Anticipated transient without scram Bypass event (loss of coolant accident in interfacing system or steam generator tube rupture) Reactivity accidents (e.g. homogenous or heterogenous dilution, accidental withdrawal or ejection of control rods)
Reactor coolant system status	Reactor coolant system pressure at core damage: — High (relief valves are challenged) — Medium (above low pressure coolant injection head) — Low (including method of depressurization) Status of safety relief valves Reactor coolant system integrity (shutdown states): — Vessel head removed — Nozzle dams installed — Safety valves removed — Vent open Reactor coolant system coolant inventory (shutdown states): — Full power inventory — Flooded refuelling cavity — Mid-loop operations in a pressurized water reactor State of fuel in the reactor for decay heat: — Operating power level — Pre or post refuelling — 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)	Injection fails to start (no injection, early damage) Coolant injection initially successful, but recirculation cooling fails (later core damage) Emergency core cooling functionality after core damage or breach of reactor pressure vessel Steam generator cooling availability

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

Status of the containment's engineered safety features	Sprays (if any): — Operate at all times — Fail on demand — Initially operate, but fail on switchover to recirculation cooling Suppression pool (if any): — Effective at all times — Ineffective (pool drained or bypassed early) — Bypassed late Fan or water coolers (if any): — Operate at all times — Fail on demand — Fail late Venting systems: — Operate at all times — Fail on demand — Fail late Status of containment inerting systems (if any) Status of means for management of combustible gases, i.e. monitoring and control by passive means (e.g. autocatalytic recombiners) and active means (e.g. igniters) — Fully available as designed — Degraded — Fully failed Containment passive heat removal system (if any): — Available — Unavailable — In operation — Failed
Status of support systems	Electrical alternating current and direct current power Component cooling Instrument air Heating, ventilation and air conditioning Availability and accessibility of mitigating systems
Containment status	Intact and isolated at the onset of core damage Intact, but not isolated at the onset of core damage Bypassed Structural failure or enhanced leakage (with indication of size and location of leakage)[a]

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

Status of secondary containment (reactor building or enclosure building)	Intact and isolated at the onset of core damage Intact, but not isolated at the onset of core damage Structural failure or enhanced leakage[a]

[a] This includes any external hazards that might damage containment structures.

(see Table 3). One method for incorporating such missing systems into the Level 1 PSA is to develop bridge trees (referred to as 'extended event trees' in Fig. 1) that link to Level 2 PSA system models, thereby capturing important dependencies (e.g. support systems, operator performance).

5.8. 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 in paras 5.11–5.13 will be implicitly available for the Level 2 PSA model. Such an approach may allow for a reduction in the number of PDSs 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 the Level 1 PSA model will not screen out possible PDSs contributing to radioactive releases.

5.9. 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 containment status (e.g. intact and isolated, intact and not isolated, failed or bypassed) should be specified for the PDS and, for PDSs where the containment is bypassed, the type and size of the bypass (e.g. loss of coolant accident in interfacing systems, steam generator tube rupture) should be specified. 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 should be specified for the PDS. For PDSs with intact containment, an accident progression event tree analysis should be performed. For PDSs where the containment is failed, not isolated 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 additional damage to the containment that increases the releases.

5.10. The characteristics specified for the PDSs are generally left to the discretion of the analyst. Examples of characteristics are given in paras 5.11–5.13. It should be noted that the level of detail of characteristics needed to define the PDSs depends on the approach used for the development of Level 1 PSA and Level 2 PSA (see para. 2.6). In either approach, the definition and selection of characteristics specified for the PDSs should be justified.

Plant damage states without containment bypass

5.11. In specifying PDSs without containment bypass, account should be taken of the equipment and system failures or scenario features, including operator actions 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, the following might need to be taken into account (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, emergency core cooling system) that have occurred, leading to core damage.
(c) Extent of fuel damage.
(d) The timing 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[14].
(f) The pressure in the reactor pressure vessel after the onset of core damage, which affects the possibility of temperature and pressure induced failures of the reactor coolant system (e.g. creep rupture of piping and steam generator

[14] In some cases, it is useful to group separately the sequences with high pressure at the onset of core damage but where the pressure is naturally reducing (e.g. loss of coolant accidents, small consequential leakages) and can result in low reactor coolant system pressure before the lower head failure.

tubes, thermal seizure of a safety or relief valve in the open position). The pressure is affected by the initiating event and the functionality of any depressurization system.

(g) The pressure in the reactor pressure vessel at the time of lower head failure, which is related to (but not necessarily the same as) the pressure at the onset of core damage and 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.
(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, saturated) when core damage occurs (for a boiling water reactor).
(k) The availability of the containment associated systems (e.g. containment sprays, heat removal systems and hydrogen mixing devices or recombiners).
(l) Initial and boundary conditions including the availability of alternating current and direct current power and associated recovery times.
(m) Actions by operating personnel that have been attempted and failed.

5.12. The status of the engineered safety features[15] 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 might influence processes such as cooling, the removal of radioactive material, or the mixing of combustible gases in the containment. For some reactor designs, the status of electrical power supply is important in grouping of accident sequences into PDSs because it impacts the availability of accident management systems. 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

5.13. For PDSs with containment bypass, the main consideration should be the identification of attributes that are associated with the attenuation of concentrations of radioactive material along the release pathway or that affect the

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

timing of release. This should include the type of initiating event, the status of the emergency core cooling system (including failure time), whether the leak pathway is isolable after a given time period, and 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).[16] For leaks into the auxiliary building or an equivalent building, 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

5.14. If the consideration of all factors and parameters that affect the Level 2 PSA results in an excessive number of potential PDSs, then they should be reduced to a manageable number. Two approaches could be taken. The first is to combine similar PDSs and perform a bounding analysis to select a representative sequence that characterizes the PDSs 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 that are similar with regard to accident progression. The second approach is to use a frequency cut-off as a means of screening out less important PDSs.[17] The analyst should justify that such screening does not introduce a significant underestimation of the risk calculated by the Level 2 PSA; careful evaluation is therefore necessary before 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 to the environment (see also para. 5.24). In any case, during 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. Consideration should be given to how this affects the specific objectives of the PSA, bearing in mind that for some PSA applications (e.g. risk monitors), the screening out approach could be inappropriate.

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

[17] In some 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 distinguish significant PDSs from less important PDSs.

PLANT DAMAGE STATES FOR LOW POWER AND SHUTDOWN MODES OF OPERATION

5.15. 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 reactor containment. The PDSs specified for full power conditions should be reviewed and adapted for low power and shutdown modes; direct use of the PDSs specified for Level 2 PSA for full power conditions might not be possible.

5.16. 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 hatches opened. The following additional accident sequence characteristics should be considered in specifying the PDSs for low power and shutdown (see also para. 9.34 of SSG-3 (Rev. 1) [4]):

(a) Decay heat level (based on time since shutdown from power operation);
(b) Conditions that determine the time taken to restore containment isolation and the potentially reduced effectiveness (i.e. leaktightness) of the containment during this time;
(c) Integrity of the reactor coolant system pressure boundary with the reactor vessel head removed, nozzle dams installed, safety valves removed and reactor coolant system vent open;
(d) Coolant inventory in the reactor coolant system;
(e) 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

5.17. In order to extend the scope of Level 2 PSA to include internal and external hazards such as fire, seismic hazards 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 if those aspects have not yet been taken into account in the Level 1 PSA. For example, in a Level 2 PSA for fire, the cables associated with the systems for ensuring the confinement function should be tracked to assess the impact of fire scenarios; or operator actions in the main control room may need to be failed if 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, and other mitigating systems and equipment 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 for ensuring the confinement function due to internal or external hazards could be assimilated into already defined isolation failures of systems for ensuring the confinement function.

5.18. The potential impact of hazards on the systems for ensuring the confinement function 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.[18]

5.19. The analyst may decide to credit human actions that occur before or soon after core damage in the Level 1 PSA and capture these human actions as part of an attribute of the PDSs for the Level 2 PSA (see 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.

5.20. The analysis process to be conducted in order to consider hazards and their combinations for Level 1 PSA is described in section 6 of SSG-3 (Rev. 1) [4]. This process is applicable to Level 2 PSA and it is not repeated here. For 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, such as the following:

(a) A beyond design basis earthquake resulting in a station blackout and a containment failure, possibly with consequential internal fire or flooding;
(b) A combination of external flooding and high winds that might lead to the loss of the heat sink together with the loss of off-site power;

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[18] Typical examples of impacts of 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).

(c) An aircraft crash causing a common loss of off-site power and emergency diesel generator failure that results in not only a plant transient but also an accident sequence with containment bypass and releases of radionuclides (via air or water).

5.21. In order to be widely applicable, the Level 2 PSA for hazards should be based on the Level 1 PSA covering these hazards, as described in section 6 of SSG-3 (Rev. 1).

5.22. 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 results of the Level 1 PSA for hazards. In particular, if there is a low level of knowledge associated with the Level 1 PSA results, the relevance of extending this PSA to Level 2 should be analysed with regard to safety issues, feasibility and ease of analysing insights from it.

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

5.24. Any hazards, single or combined, that were screened out from further (bounding or detailed) analysis within the Level 1 PSA should also be reassessed, consistent with paras 6.17–6.19 of SSG-3 (Rev. 1) [4], taking into account that para. 6.19 of SSG-3 (Rev. 1) [4] 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, their ability to affect the confinement function should be considered. In this context, there should be a distinction between the following:

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

5.25. Based on the identification and screening of hazards performed within Level 1 PSA, those combined hazards, for which either simple or detailed probabilistic analyses have been performed in the framework of Level 1 PSA, should be analysed within Level 2 PSA. In line with the guidance provided in section 6 of 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. Severe accident progression analysis typically consists of different groups of analyses performed in different phases of a Level 2 PSA project. Early in the project, severe accident progression analysis is used to understand the general core damage accident progression for key initiating events, providing a starting point for the Level 2 PSA analysis [19]. Subsequently, analyses are performed to support the specification of PDSs, which assists the event tree analysts in establishing which systems and accident progression features are most important for determining the plant response and, hence, need to be included in the PDS specifications or in the accident progression event tree model, as appropriate for the methods being applied in the PSA. For example, investigations may provide insights into the variation of accident progression when different numbers of injection trains are operating or into the influence of primary and secondary pressure status (in a pressurized water reactor), or they may indicate the effect of changes in the volume of water injected to containment on ex-vessel molten core behaviour. A third area where severe accident progression analyses provide support is in the assessment of specific phenomena, where accident analysis results may be used as an input to phenomenological probability calculations (see Section 9) and help to specify the timing for human action events included in the logic models. Finally, severe accident progression analyses support the grouping of accident progression event tree sequences into release categories (see Section 11), which is a process similar to the definition of PDSs, and the quantitative characterization of the 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, such as accident timing, main phenomena, conditions for manual actions and automatic actions, effects of systems actuation and 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 and equipment reliability analysis (see Section 8), in the development of accident progression event trees (see Section 9) and in support of source term analysis (see Section 10).

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

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

6.5. Severe accident progression analysis should be performed by teams with experience in the application of severe accident codes; if necessary, appropriate training should be included in the project (see paras 3.19–3.22 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 identified by the Level 1 PSA and grouped in specific PDSs should provide key information such as fuel uncovering 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 (see para. 5.14 and footnote 17) should be mapped to a specific representative calculation; some calculations can represent more than one PDS, if justified. Calculations could also be performed for those PDSs that have a low occurrence frequency but 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 for those with severe consequences, a sufficiently wide range of information is usually 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 assessing reactivity accident scenarios resulting in prompt criticality accidents leading to reactor core damage and potential damage to the containment integrity.

6.10. Severe accident progression analysis for 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. To meet Requirement 18 of GSR Part 4 (Rev. 1) [2], decisions on the computer code(s) used to perform detailed analyses and the number of calculations to be made should be based on the objective of the Level 2 PSA. In making these decisions, the following should also be taken into account:

(a) The code(s) should be capable of modelling most of the initiating events that were 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 code(s) should be verified and validated against the severe accident phenomena analysed by them, and a validation matrix should be available.
(d) Validation and benchmarking, and the associated documentation, should be sufficient to support the necessary severe accident progression analyses (see also Ref. [20]).

6.12. The analysts should be adequately trained in the use of the code(s) to be applied and should be aware of the technical limitations and weaknesses of the selected code(s) (see para. 3.21).

6.13. The analyses of severe accidents should first cover key sequences for each PDS that leads either to a successful controlled state of the plant, where sufficient safety systems or safety features have operated correctly so that all of the required safety functions necessary to cope with the sequences have been fulfilled, or to filtered containment venting (if provided) or to a degraded state[19] with one or

[19] A 'degraded state' is considered to be a state following a severe accident, in which the safety functions performed by the containment and its associated systems are affected.

several containment failures. Second, the remaining sequences in the accident progression event tree should be quantified for confirmation purposes.

6.14. To support 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 should be performed. Integral analyses start with the initiating event and end in accordance with appropriate criteria, depending on the purpose of the analysis. Examples of criteria for when to terminate analyses are (a) when the cumulative release of radionuclides into the environment has stabilized, (b) after corium stabilization (in-vessel or ex-vessel), or (c) after a predetermined mission time has elapsed. The analysis of individual phenomena should be supported by severe accident progression analyses, as necessary (see also Section 10). Some examples of individual phenomena for water cooled reactors are as follows:

(a) Structural–mechanical behaviour of components of the reactor coolant system in the event of high-pressure severe accident scenarios;
(b) Interaction of the core, core structures and corium with coolant inside and outside the reactor pressure vessel (e.g. quenching, steam production, steam explosion, hydrogen generation and induced effects on the plant);
(c) Ex-vessel cooling of the reactor pressure vessel for in-vessel melt retention;
(d) High pressure melt ejection and direct containment heating;
(e) Hydrogen and carbon monoxide generation by molten core concrete interactions, flow and distribution in the reactor containment and mitigation means to cope with combustion behaviour;
(f) Ex-vessel corium cooling;
(g) Criticality accident effects;
(h) Containment pressurization.

6.15. In general, the analyses should be performed in a best estimate manner with regard to applied codes, models, model parameters and boundary conditions. Conservative assumptions, which are in common use for design basis accident analysis, might not be useful or appropriate in severe accident progression analyses for Level 2 PSA because, for example, conservative assumptions might distort the results and the risk insights and consequently might lead to deviation from optimal severe accident management strategies.

6.16. Accident management measures for both prevention of core damage and mitigation should be considered in the severe accident progression analyses with realistic timing for human actions.

6.17. All severe accident progression analyses (including descriptions of input decks, boundary conditions, assumptions and results) should be part of the Level 2 PSA documentation in order to provide the justification for 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

6.18. Analysis of the severe accidents in SFPs that are 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 recommendations on severe accident progression analysis in the SFP.

6.19. Depending on the SFP location (i.e. inside the reactor containment, outside the reactor containment but inside the reactor building, or outside the reactor building), severe accident progression analysis should provide information on the interactions between the reactor and the SFP: there may be mechanisms through which a reactor accident can induce a SFP accident and vice versa. The outcome of this analysis may be additional accident scenarios (involving both the reactor and the SFP) being added to the Level 2 PSA.

6.20. If the SFP is inside the reactor containment and the joint accident progression event tree sequences involving the reactor and the SFP are not demonstrated to be negligible, severe accident progression analysis should be performed to show the combined impact of reactor and SFP accidents on conditions in the containment (e.g. pressure, temperature, corium spreading, inflammable gas, steam production in the containment, acceleration of evaporation of SFP water) and on radioactive releases.

6.21. If the SFP is located inside the reactor building but outside the reactor containment, then the severe accident progression analyses should consider the impact of severe accident progression inside the reactor building and its effects on the SFP and on radioactive 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 SFP might be possible, thus allowing the use of common systems and common severe accident management strategies for ensuring decay heat loads in each of these locations and for possible fuel assembly handling.
(e) In water cooled reactors, when the reactor pressure vessel head is closed (and provided that the decay heat level and operating configuration are similar to those for full power), core melt accident phenomena might be considered very similar to the sequences that occur in full power mode. When the reactor pressure vessel head is open, 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 some reactor designs, during the majority of shutdown states when the reactor pressure vessel is open, there is a water pool connecting both the reactor and the spent fuel pool.

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

IDENTIFICATION OF SOURCES OF UNCERTAINTY

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

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 and sensitivity analyses should be derived. Care should be taken if including parameters such as correlation coefficients and parameters for modelling the phenomenology of severe accidents in the corresponding computer codes, which are established as part of the computer code validation procedure. The list of parameters should be established on the basis of a systematic process and good practices should be followed in performing severe accident 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 SEVERE ACCIDENT PROGRESSION FOR 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 Relocation of corium into lower head of reactor vessel Corium stratification inside lower plenum of reactor vessel (metallic/oxidized layers, focusing effect for thermal flux) Ex-vessel cooling (for in-vessel retention)
In-vessel forced or natural circulation	Circulation flows in reactor coolant system loops influenced by the presence of water in cold leg (direct or counter-current steam flows) Competing mechanisms of degradation and failure of reactor coolant pump seal

TABLE 4. EXAMPLES OF AREAS OF UNCERTAINTY RELEVANT TO SEVERE ACCIDENT PROGRESSION FOR WATER COOLED REACTORS (cont.)

Type of severe accident event	Related phenomena
In-vessel corium–water interactions (energetic and non-energetic)	Effect of in-vessel water injection (quenching) after recovery of systems or components: pressure peak, hydrogen generation, fuel cooling depending on core degradation progression and water flow rate Potential for terminating in-vessel fuel degradation Recriticality Steam explosion, high pressure failure of the reactor pressure vessel Release and dispersion of radioactive material
Failure of primary circuit	Break size Break location
Failure mechanisms of reactor pressure vessel and loop boundaries	Melt penetration and cooling within vessel head penetrations Local or global failure of lower head of reactor pressure vessel: mechanical (creep) or melting failure Impact of ex-vessel cooling 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 flaws
High pressure melt ejection and/or direct containment heating	Trapping of melt debris on containment structures Heat release by zirconium oxidation and additional hydrogen production Debris transport from the vessel/cavity to the containment atmosphere 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) Quasi-static increase in containment pressure (steam spike) Local dynamic loads to containment from steam explosion (reactor cavity) and possible damage to structures Releases of radioactive material

TABLE 4. EXAMPLES OF AREAS OF UNCERTAINTY RELEVANT TO SEVERE ACCIDENT PROGRESSION FOR WATER COOLED REACTORS (cont.)

Type of severe accident event	Related phenomena
Core–concrete interactions	Erosion of containment structure (basemat) by debris Generation of non-condensable and/or combustible gas (e.g. CO, CO_2 and H_2) Lateral/axial spreading of debris and potential for contact with containment boundary Corium spreading Corium coolability Effects of presence of metal within the melt or within the concrete, melt stratification (metallic/oxidized layers) Releases of radioactive material
Hydrogen and carbon monoxide combustion	Mixing and/or stratification of flammable gas in containment atmosphere Local increase of concentrations (e.g. due to strengthened steam condensation in cold containment regions) Steam or nitrogen inerting Hydrogen and carbon monoxide recombination, ignition time and combustion, flame acceleration and transition from deflagration to detonation Heat losses to structures Containment structure response to combustion pressure wave (e.g. open doors or blow-out panels, displacement of water pools) 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 systems associated with containment. As such, the applicability of this section depends on the reactor technology and design. In the 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 address the following:

(a) The capability of the containment to maintain its leaktightness and structural integrity under internal loads (see 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) (see 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 (see paras 7.17–7.20 and 9.3–9.6).

7.3. Containment integrity might also be impacted by hazards. For instance, high level seismic events might directly cause a loss of containment leaktightness. Therefore, a review should be conducted of the impacts caused by hazards within the scope of PSA in order 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 (see paras 5.17–5.25).

ANALYSIS OF REACTOR CONTAINMENT PERFORMANCE

Analysis of containment performance under internal loads

7.4. In general, the goal of the containment structural analysis with regard to internal loads is to assess the probability of containment failure as a function of pressure and/or temperature under severe accident conditions[20], known as

[20] Examples of phenomena leading to overpressurization loads could be combustion of combustible gases such as hydrogen combustion (i.e. deflagration and detonation) and carbon monoxide combustion.

a fragility curve[21] or a fragility (hyper)surface[22] (see Ref. [21] for examples). Typically, material properties vary with containment structure temperature; consequently, pressure driven failure modes will be affected by temperature conditions. Usually, an enveloping temperature is chosen (on the basis of severe accident progression analyses) and the overpressure analysis is then performed assuming those temperature conditions. The development of containment fragility curves should include consideration of pressure driven and temperature driven failure modes that are applicable to the containment design under consideration in the Level 2 PSA. This analysis should include an identification of potential failure modes and their respective locations.

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 specified for containment leak rates (e.g. threshold values 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 might not take into account 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 should be collected. The following are examples of important features of the structural design of the containment and containment penetrations that should be taken into account for water cooled reactors:

(a) Containment material:
   (i) Steel;
   (ii) Concrete:
      — Prestressed;
      — Post-tensioned;

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[21] A fragility curve represents the probability of containment failure as a function of one variable, such as pressure or temperature.

[22] A fragility surface represents the probability of containment failure as a function of two or more variables together, such as pressure and temperature.

— Reinforced;
— With or without steel liner;
— With or without resin (e.g. epoxy) liner.

(b) Containment penetrations:
  (i) Equipment hatch(es);
  (ii) Personnel hatch(es);
  (iii) Piping penetrations;
  (iv) Electrical penetrations;
  (v) Atmosphere purge line(s);
  (vi) Vent line(s).

(c) Other aspects:
  (i) Geometrical shape of containment (sphere, cylinder, rectilinear);
  (ii) Geometric details of the containment structure, penetrations and hatches;
  (iii) Geometrical discontinuities (e.g. transition from cylindrical shell to top head and basemat);
  (iv) Steel or resin (e.g. epoxy) liner anchorages;
  (v) Details of any reinforcements around penetrations;
  (vi) Materials and their properties;
  (vii) 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 achieved by making plant specific structural calculations that account for the containment design features listed in para. 7.6. However, depending on the scope of the Level 2 PSA, use can be made of existing calculations for plants with 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, the applicability of the existing structural response analyses to the plant under consideration and the 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 in accordance with para. 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 use of the 'threshold model' and the 'leak before break model'. The threshold model defines a threshold pressure, with 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 lower than or equal to the leakage rate associated with a leak failure, containment pressurization will be stopped if such a failure occurs, thereby 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. Best practice is to include pressure ramp rates, localized heat loads and localized or global dynamic pressure loads. Dynamic pressure loads may 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. equipment hatches, personnel hatches) and singularity zones can be relative weak points 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 equipment hatch closure system is to be studied under severe accident conditions [22, 23].

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 [23] and 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 also be affected by failure mechanisms induced by severe accident phenomena. Examples of phenomena to consider are induced fires (e.g. graphite fires), steam explosions (e.g. instantaneous vaporization of water induced by its contact with molten corium), chemical attacks (e.g. chemical reactions affecting containment structure integrity) and direct contact between molten core debris and the containment structure. Recommendations related to the consequences of molten core debris and containment structures for the containment integrity analysis are provided in paras 7.13–7.16.

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 that extensive erosion is possible as a result of such interactions (see Section 6). This may be of particular concern for the response of the containment basemat but also, depending on the plant design, the response of the 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 are 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 a complete or partial penetration by core debris should be examined if calculations of severe accident progression suggest such levels of erosion are possible.

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 as part of the analysis of reactor containment performance. All of the containment penetrations should be modelled, or a careful justification should be provided to justify that some penetrations can be screened out. Screening criteria may be applied in order to focus on the penetrations that are most likely to result in significant releases. For instance, containment isolation might not be modelled for normally closed lines if the isolation valves would not be opened during the accident (e.g. due to the initiating event or type A human failure event (see para. 8.1)) or for closed loop systems inside the containment, if closed loop integrity would not be threatened during the accident. Any plant operating feedback regarding containment isolation valve leakages should be taken into account.

7.18. Possible failure modes of valves should be taken into account, such as failure to close and spurious opening. Leaks 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 (i.e. release from the core to the environment without being able to credit containment) should be assessed through interface with the Level 1 PSA (see Section 5) or through accident progression event tree logic (see Section 9). The bypass paths should be identified by a rigorous search of systems that are located outside the containment and linked to reactor coolant loops or containment atmosphere. In general, containment bypass might result either from an initiating event (e.g. steam generator tube rupture, loss of coolant accident in interfacing system), from an induced event due to accident conditions (e.g. severe accident induced steam generator tube rupture for a pressurized water reactor) or from a failure (e.g. leak, incorrect circuit configuration) 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 open equipment hatch. If such penetrations are present, the assessment of containment internal loading and the effects on concrete structures due to molten core debris interactions are 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 section are applicable.

Analysis of containment isolation failure during shutdown

7.22. With regard to containment isolation failure, particular attention should be paid to the shutdown mode of operation, during which the following might apply:

(a) Personnel airlocks or equipment hatches might 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 an equipment hatch, the analyst should take into account aspects such as whether the closure is requested by emergency operating procedures, the time taken to perform the closure and the dependencies (e.g. power supply) involved.
(b) Isolation lines that are normally closed at power might be open and might 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 the following:

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

7.24. Material variability and modelling uncertainty can be determined as part of the structural capacity assessment by techniques for uncertainty quantification and propagation in order to establish a failure pressure and/or temperature distribution function. Statistical feedback based on test samples from construction sites may be useful for assessing material variabilities. Benchmarks from mock-ups or feedback experience from pressure tests (if available) may be useful for assessing modelling uncertainty. One example for assessing modelling uncertainty is provided in Ref. [21]. Experiments are supported by analytical predictions of containment failure pressure; these analytical predictions are useful for transferring knowledge at the reactor scale (see, e.g., Ref. [22]). Alternatively, expert judgement supported by simple analysis could 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. Individual fragility curves are also needed to calculate 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. 7.23).

Characterization of uncertainties related to concrete structure erosion by molten core debris

7.26. Uncertainties associated with the resistance of concrete structures to extensive erosion by molten core debris should be assessed. In a similar way, as described in para. 7.23, uncertainties are the result of several factors, including material variability and modelling uncertainty.

7.27. Uncertainties affecting the development of the molten core–concrete interaction 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 type of concrete used for the basemat construction.

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

These uncertainties can be assessed, for example by using Ref. [24], which comprehensively summarizes various topics related to these interactions, such as available experiments, plant application, simulation tools and models, and uncertainties.

Characterization of uncertainties related to containment isolation failure and containment bypass

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

(a) Data used (e.g. component reliability data, unavailability due to maintenance, opening duration of large penetrations such as equipment hatches or personnel airlocks);
(b) Human reliability analysis (see Section 8);
(c) Severe accident phenomena modelling (see Section 6).

8. HUMAN AND EQUIPMENT RELIABILITY ANALYSIS

HUMAN RELIABILITY ANALYSIS

8.1. Human failure events can be classified in the same way for Level 2 PSA as for Level 1 PSA (see paras 5.97–5.122 of SSG-3 (Rev. 1) [4]):

(a) Type A human failure events are those that occur before the initiating event and that have the potential to lead to the failure or unavailability of credited systems. 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 could lead to an initiating event. These events are included in the Level 1 PSA but are not relevant to Level 2 PSA.
(c) Type C human failure events are those that might occur during critical actions taken by operating personnel after the occurrence of an initiating event. Identifying and analysing Type C human failure events is the main task performed in human reliability analysis for Level 2 PSA. Paragraphs 8.2–8.12 provide recommendations on 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 release of radionuclides should be modelled in Level 2 PSA. In general, these actions are identified in the following:

(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 criteria for entering severe accident management guidelines).
(b) Severe accident management guidelines, including:
    (i) Operator actions that can be implemented without the support and/or approval of response organizations;
    (ii) Operator actions that can be implemented only with the support and/or approval of response organizations (or technical support centre).
(c) Strategies and guidelines for the deployment of non-permanent equipment or additional strategies not considered in emergency operating procedures or severe accident management guidelines, if such strategies and guidelines are established.

8.3. These human actions are most often taken by operators (in the control room) or by field workers (outside the control room). However, some actions can also be taken by external teams (i.e. external to the plant organization) specially trained to handle severe accidents (e.g. an external response organization that transports non-permanent equipment from another (often distant) location to the site to mitigate the event).

8.4. Some operator actions considered in the Level 1 PSA human reliability analysis may be considered for applicability in the Level 2 PSA. Some actions may be considered failed in the context of Level 1 PSA but could become feasible in the Level 2 PSA if an extended time window is available. This is because the criteria for core damage considered in Level 1 PSA are more restrictive than the criteria applied in Level 2 PSA for arresting the accident progression.

8.5. Human actions (e.g. manual opening of pressure operated relief valves) that are 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 impact 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, such as 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 analysis 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 combinations of hazards 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. Specificities in the human reliability analysis for a Level 2 PSA should be taken into account, such as the following:

(a) How human actions are prescribed. Depending on the organization in place to deal with a severe accident, some actions may be taken independently by the plant staff, while others need to be approved or directed by the technical support centre. In the latter case, the technical support centre needs to be fully operational, and good coordination with plant staff is needed.
(b) The lack of specific instructions in severe accident management guidelines as compared with emergency operating procedures. The lack of specific instructions 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 levels and/or high temperature in certain rooms, potentially degraded or absent lighting) should be taken into account, especially when the necessary action is taken locally, because these conditions might affect human reliability.

8.8. The Appendix provides more detailed information about performing human reliability analysis for a Level 2 PSA. Human reliability analysis in the context of a Level 2 PSA for multi-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 analysis (see para. 5.110 of SSG-3 (Rev. 1) [4]). In particular, the following should be taken into consideration:

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

8.10. Dependencies between operator actions should be assessed and incorporated into the Level 2 PSA model. This includes dependencies between the human actions credited in the Level 2 PSA and those credited in the Level 1 PSA and Level 2 PSA. The degree of dependency can be influenced in particular by the organization and procedures that are implemented at the nuclear power plant, the context of each human action (i.e. preventing core melt or mitigating severe accident progression), the delay between human actions, and whether the human actions are performed by the same people or using the same equipment.

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 an 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 should be used to identify or improve severe accident management actions, as explained in Section 15.

EQUIPMENT RELIABILITY ANALYSIS

8.13. Equipment reliability, including common cause failures, is usually modelled in a Level 2 PSA 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 in the extended event trees (i.e. bridge trees; see para. 5.7).

Recommendations related to equipment reliability analysis in Level 2 PSA models are provided in paras 8.14–8.17.

8.14. Assessment of the reliability of equipment credited within the Level 2 PSA should consider the periodic testing and maintenance practices or planned maintenance procedures for installed equipment, as well as the deployment, installation, start-up and operation procedures for non-permanent equipment. 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 for 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 might include high levels of temperature, pressure, humidity and radiation in the containment and auxiliary buildings. Examples of adverse conditions that could affect equipment reliability are energetic events (e.g. short term temperature and pressure spikes, impulse loadings from detonations or steam explosions) or high radiation levels that impact specific SSCs (e.g. electronic instrumentation, rubber gaskets vulnerable to high radiation levels).

8.16. Repair actions should be credited in a 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 (a) it is possible to diagnose the failure, (b) the spare parts and repairing personnel are in place, (c) the environmental and work conditions needed for access and for performing the repair are given or can be ensured, and (d) the time window is sufficiently long to credibly assume the possibility for repair, including the time needed to bring spare parts and maintenance personnel to the plant. Reliable data should be used to assess credible probabilities of repairing components and systems. For components that are not reparable after a severe accident but are needed continuously after core melt (e.g. for corium cooling), the failure probability assessment should integrate the mission time for corium cooling. The failure modelling for different time windows could be discretized to consider different consequences as a function of the instant of failure.

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

8.18. The mission time of each SSC credited in the Level 2 PSA accident progression event tree should be defined in accordance with the role of the SSC during the severe accident progression until the plant reaches a controlled state (see para. 5.5). This mission time is usually derived from the severe accident progression analyses. The SSC mission times for Level 2 PSA that are defined in this way may be different from or the same as those used in Level 1 PSA. The analyst should verify that the reliability data is suitable for use with the defined SSC mission time.

IDENTIFICATION OF SOURCES OF UNCERTAINTY IN RELIABILITY ANALYSIS

Uncertainties in human reliability analysis

8.19. Uncertainties in human reliability analysis should be addressed in the same way as for human reliability analysis credited in Level 1 PSA.

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

8.21. Many human reliability analysis methods use a simplified approach to assessing 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.

Uncertainties in equipment reliability analysis

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

8.23. Uncertainties in equipment qualification or survivability with regard to severe accident conditions should be addressed in Level 2 PSA, taking into account areas of uncertainty related to both the evaluation of specific Level 2 PSA environmental conditions and 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 or a separated approach, as described in para. 2.6. The Level 2 PSA analyst should be trained in both approaches and the associated computer codes and should be aware of their limitations and the requirements imposed by the use of the approach and the computer codes chosen. More information can be found in Ref. [5].

9.2. In Level 2 PSAs, event trees are used to delineate the sequence of events and severe accident phenomena that challenge containment integrity after the onset of core damage and the successive barriers to a radioactive release. Event trees provide a structured approach to the systematic evaluation of the capability of a plant to cope with severe accidents. The use of event trees in PSA 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 effects on the confinement of radioactive products or on radioactive releases to the environment.

STRUCTURE OF ACCIDENT PROGRESSION EVENT TREES AND NODAL QUESTIONS

9.3. Nodal questions (also referred to as top events) in an accident progression event tree 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 accidents, challenges to containment integrity and associated barriers to a radioactive release, the physical containment conditions at the time of a radioactive release, and the eventual magnitude of the release to the environment.[23] The nodal questions of the accident progression event tree are specific to plant type, meaning that issues of importance to severe accident behaviour in one type of reactor and/or containment system might not be important to others. The complexity of the accident progression event tree depends on the scope and objectives of the Level 2 PSA.

[23] Nodal questions may also address issues and actions relating to severe accident management.

9.4. As the list of events and processes can be extensive, 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 [25][24]. 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. The following are examples of phases for water cooled reactors:

(a) Phase 1: Immediate response by the plant to the PDS caused by the initiating event during 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 4: Ex-vessel phenomena and events:
    (i) Phase 4a: Short term ex-vessel phenomena and events.
    (ii) Phase 4b: Long term ex-vessel phenomena and events.

9.6. Phase 1 is the initial period of in-vessel core damage (e.g. fuel rod heating up above the temperature criterion for fuel integrity, generalized oxidation of fuel cladding, melting of startup control rods). Phase 2 typically starts at the time of core melting and relocation to the lower plenum of the reactor pressure vessel (e.g. formation of debris from melting of fuel cladding, reactor pressure vessel

[24] Reference [25] describes all of the technical elements necessary for developing event trees capable of assessing large early release frequency. In some States, the large early release frequency metric is used in regulatory risk informed decision making.

internals and fuel, relocation of debris and melt in the lower plenum of the reactor pressure vessel). Phase 3 starts at the time of reactor pressure vessel failure and covers the phenomena that occur immediately after (e.g. direct containment heating, ex-vessel steam explosion). Phase 4 covers both the short and the long term ex-vessel phenomena and events after the reactor pressure vessel rupture. Phase 4a covers the period up to a few hours after failure of the reactor pressure vessel and addresses immediate ex-vessel molten core behaviour (e.g. stabilization of ex-vessel melt, onset of core–concrete interaction, human actions, equipment behaviour). Phase 4b covers the long term, starting from a few hours after failure of the reactor pressure vessel, and addresses challenges arising from ex-vessel melt behaviour (e.g. pressurization due to the generation of non-condensable gases during core–concrete interaction, combustion phenomena, pressurization due to ongoing steam generation, human actions, equipment behaviour) (see also paras 7.1–7.22 and Section 8). Typical nodal questions for an accident progression event tree used in separated models for a typical pressurized water reactor with a large, dry containment are provided in Table 5. These or similar nodal questions should also form the basis for the accident progression event tree when the integrated model is used. The nodal questions presented in Table 5 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 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
Phase 1: Immediate response of the plant to the PDS caused by the initiating event during the 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	PDS; accident progression
1	Is the containment isolated?	None	PDS

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
2	What is the fraction of the PDS with alternating current power available?	None	PDS
3	What is the mechanical status of sprays in the very early time frame?	None	PDS
4	What is the mechanical status of fans in the very early time frame?	None	PDS
5	Is the reactor coolant system depressurized manually in the very early time frame?	2	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	PDS
9	Are containment sprays actuated in the very early time frame?	3, 6, 8	Accident progression

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	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 (a) the fission product releases (resuspension), (b) the containment integrity or (c) 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	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 in-vessel core damage up to 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; 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 is the mode of reactor pressure vessel failure and the process of core debris ejection?	5, 6, 7, 14, 15	Accident progression

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
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; 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 heating up of direct containment 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	Is the filtered vent system actuated at breach of the reactor pressure vessel?	1, 11, 13, 15, 16, 19, 20, 21	Accident progression
	Phase 4: Ex-vessel phenomena and events		
	Phase 4a: Short term ex-vessel phenomena and events		
23	Is alternating current power restored or maintained in a short time frame?	8	PDS

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
24	Are sprays actuated or are they restored to operate in a short time frame?	9, 23	PDS; accident progression
25	Are fan coolers actuated or are they restored to operate in a short time frame?	4, 23	PDS
26	Is core debris in a coolable configuration outside the vessel?	15, 16, 17, 18, 19	Design features of the core catcher; accident progression
27	Is a containment heat removal system in operation or restored in a 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 a longer time frame?	23	PDS
29	Are sprays actuated or do they continue to operate in the late time frame?	24, 28	PDS; accident progression
30	Are fan coolers actuated or do they continue to operate in the late time frame?	25, 28	PDS
31	What is the status of fans and containment sprays in the late time frame?	29, 30	Summary type question

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
32	Does hydrogen combustion occur in the late time frame, and what is the induced pressure peak in the containment? (Does it affect (a) the fission product releases (resuspension), (b) the containment integrity, or (c) specific equipment in the containment (local effects)?)	10, 20, 31	Accident progression
33	Is the filter vent system actuated 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
36	Does containment failure (e.g. slow overpressurization, hydrogen combustion) occur in the late time frame?	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. Methodology for the estimation of nodal probabilities using both the threshold approach (see para. 9.12) and convolution approach (see para. 9.13) may be provided in the documentation attached to the Level 2 PSA. Account should be taken of issues that could affect the analyst's ability to predict the progression of severe accidents and assign uncertainties, including limitations in knowledge regarding aspects of severe accident phenomena, model completeness, fidelity and validation of available computer codes, and applicability of available experimental data to full scale reactor conditions. Example methods for dealing with such uncertainties and the use of expert judgement and expert elicitations can be found in Refs [26–33].

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 in accordance with the governing phenomena [34, 35]. 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 they 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. In this context, linked event trees, fault trees (see, e.g., Ref. [36]), user defined functions and other methods have been used for developing and quantifying accident progression event trees [37].

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. The Level 2 PSA model typically involves events of different natures, such as system operation, human action and response of containment or components to severe accident phenomena. Recommendations for assessing human action failure and system equipment failure in the context of a Level 2 PSA are provided in Section 8. Paragraphs 9.10–9.14 provide recommendations on determining probabilities associated with the response of the containment or components 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 the quantification of probabilities can include the following:

(a) Basic physical principle calculations (e.g. cooling water flow rate compared to core decay heat) 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 paras 9.12–9.14. Reference [38] 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 that study was published in 1990, 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 [5, 39, 40]. Developments in water cooled reactors have taken place in a number of areas, such as the following:

(a) In-vessel steam explosions (alpha mode containment failure) [34];
(b) Direct containment heating [41, 42];
(c) Failure of the lower head of the reactor pressure vessel [43, 44];
(d) Flame acceleration and the transition from deflagration to detonation [45];
(e) Thermally induced steam generator tube rupture and hot leg failure [46];
(f) Recovery of partially degraded cores [47, 48].

Threshold approach

9.12. 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

9.13. 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 reached 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 (i.e. the subjective probability for) failure. In this case, the consistency of the resulting probability values is dependent on consistent assignment of distribution parameters (i.e. median values, deviations about the median, choice of distribution type and limits).

9.14. Both the threshold approach and the convolution approach can be applied either independently or in combination in the PSA. In any case, to ensure 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

9.15. 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 normally be done using a computerized tool because of the necessity for efficiently handling a large amount of information. The particular way that this is done depends on the software used for quantification of the accident progression event tree, but it can involve post-processing 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.

9.16. The end states of the accident progression event tree grouped in a release category are expected to have similar radioactive release characteristics and off-site consequences. The source term analysis performed for the group therefore characterizes the entire set of end states within the group and reduces the amount of source term analysis that needs to be performed (see Section 10).

10. SOURCE TERM ANALYSIS FOR SEVERE ACCIDENTS

10.1. This section provides recommendations on release category specification and source term analysis. For a PSA with lower objectives, 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 in accordance with the objectives of the PSA.

10.2. 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 performed only for a representative severe accident scenario of each release category. The preliminary list of representative severe accident scenarios should be based on severe accident scenarios established for identified PDSs (see para. 6.8). The choice of representative scenarios should be justified. It is good practice to perform sensitivity studies for the choice of representative scenarios.

10.3. Source term analysis for severe accidents consists of two subtasks: the definition of release categories and the calculation of radioactive releases for each of these categories. In the first subtask, 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 radioactive release that will arise for accident progression event tree sequences (see paras 10.5 and 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 radioactive release (see para. 10.7).[25] In the second subtask, to calculate the radioactive 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).

10.4. Source term analysis in Level 2 PSA should involve the following steps:

(1) Defining the release categories;
(2) Grouping the end states of the accident progression event tree into the release categories;
(3) Performing source term analysis for the release categories.

SPECIFYING AND GROUPING RELEASE CATEGORIES

10.5. Taking into account the reactor technology, the analyst should consider events 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 (e.g. cooling water) and the efficiency of physical mechanisms for cooling molten core material, considering the depth and composition of ex-vessel core debris;
(e) The availability of credited systems able to reduce radioactive releases (e.g. containment spray system, filtered containment venting system, suppression pool, ice condensers);
(f) The retention mechanisms for radioactive material (e.g. pool scrubbing, retention in pipes, filters).

[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.

TABLE 6. 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 (i.e. at the bypass of the containment) Early (i.e. during in-vessel core damage) Intermediate (i.e. immediately following breach of the reactor pressure vessel) Late (i.e. several hours after breach of the reactor pressure vessel)	Typical attributes of Level 2 PSA
Pressure of reactor pressure vessel during core damage	High (near nominal) Low (depressurized)	
Containment pressure	High Low (depressurized)	
Modes or mechanisms of containment leakage (associated with a time frame)	Design basis conditions leakage Beyond design basis conditions leakage Catastrophic rupture of containment Containment bypass: — Loss of coolant accident in interfacing system — Steam generator tube/tubes or header rupture Open containment isolation valves Open material hatch access Basemat penetration	
Active engineered features providing capture mechanisms for radioactive material	Sprays Fan coolers Filtered vents Others (e.g. water management, reactor coolant system depressurization)	

TABLE 6. EXAMPLES OF TYPICAL ATTRIBUTES USED FOR THE SPECIFICATION OF RELEASE CATEGORIES FOR WATER COOLED REACTORS (cont.)

Release attributes	Variations	
Passive engineered features providing capture mechanisms for radioactive material	Secondary containments Reactor buildings Suppression pools Overlying water pools	
Passive containment coolers Ice beds 'Tortuous' release pathways Submerged release pathways Alkaline materials		
Phenomena	Molten core–concrete interaction Deposition of aerosols Resuspension of deposed aerosols by energetic phenomena Specificities for chemical process (e.g. iodine, ruthenium)	Additional attribute for linking to Level 3 PSA
Time elapsed since the start of the severe accident[a]	Short (for pressurized water reactor typically less than 2 hours) Medium (for pressurized water reactor typically between 2 and 10 hours) Long (for pressurized water reactor typically more than 10 hours)	
Location of release	Ground level Elevated	
Energy of release	Low (minimal buoyancy in ex-plant atmosphere) Energetic (highly buoyant)	
Release rate	Rapid 'puff' release Slow continuous release Multiple plumes	
Containment failure size	Sizes proposed in square metres	
Source term	Amount and composition of different radioactive nuclides or nuclide groups Duration of the release (e.g. release occurring during number of hours)	

TABLE 6. EXAMPLES OF TYPICAL ATTRIBUTES USED FOR THE SPECIFICATION OF RELEASE CATEGORIES FOR WATER COOLED REACTORS (cont.)

Release attributes	Variations

[a] Time in relation to the site specific emergency (evacuation) plan.

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 for water cooled reactors are shown in Table 6. The release of radioactive material to the environment is a function of these attributes.

10.7. The set of attributes in Table 6 can be used to specify the set of release categories used for 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, which 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 10.16–10.21). This grouping process can be less condensed if a mechanistic source term code is used for the source term analysis (see paras 10.22–10.26).

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 its importance in the global results.

10.9. For slow containment overpressurization sequences, the analyst should distinguish between containment leak and containment break, as a leak might prevent a major containment failure and subsequently limit the amount of released fission products (see para. 7.8).

10.10. The grouping of accident progression sequences in release categories often necessitates the application of some assumptions and simplifications that might introduce additional uncertainties. Special care should be taken to keep track of any assumptions and simplifications so that the additional uncertainties are not overlooked during the uncertainty analysis.

SOURCE TERM ANALYSIS APPROACHES

10.11. 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 the 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. 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.

10.12. Depending on the reactor technology, a combination of the following approaches could be used for the source term calculations:

(a) Applying one of the integral computer codes 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 section on dedicated codes in Annex I) to a large number of accident scenarios;
(c) Applying detailed models or bounding estimates of source terms or transposition from another nuclear power plant to obtain preliminary results, for example during the design phase.

The methodology and the justification of the selection of the approach followed should be provided as part of the documentation of the Level 2 PSA.

10.13. The extent to which source term analysis needs to be performed depends on the objectives and intended applications of the Level 2 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 (see also Refs [49, 50]). The justification of the adequacy should be developed and documented. The analysis of off-site consequences necessitates 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) [51].

10.14. 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, including that different release categories may have the same source term (i.e. same amount and composition).[26] However, it is important to specify these attributes at the beginning of the Level 2 PSA project (see paras 2.17–2.19).

Source term analysis with a plant specific approach

10.15. Plant specific source term analysis can be performed for each of the release categories. Such an analysis can be conducted using an integral computer code or a fast-running (also referred to as 'dedicated') computer code (see Annex I).

Source term analysis with an integral computer code

10.16. 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 such as the 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]

10.17. In the source term analysis, all of the processes that affect the release and transport of radioactive material inside the containment and in adjacent buildings should be modelled. These processes include the following:

(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, revaporization, condensation and re-entrainment (e.g. as a result of energetic phenomena, chemical reactions or mechanical effects).

[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 a Level 3 PSA will be performed.

[27] In some Level 2 PSAs, parametric source term models have been developed on the basis of calculations performed with codes such as MAAP [52] or MELCOR [53]. This approach enables the uncertainties in the source term parameters to be combined with the integrated process for uncertainty assessment and uncertainty propagation.

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

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

10.20. Source term calculations with integral computer codes for severe accident progression analysis generally consider group categories of radioactive elements or chemical compounds rather than individual radioisotopes [54, 55]. This simplification is necessary to reduce the hundreds of radioisotopes 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. Group categories also take into account similarities in the chemical affinity of 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). Typical group categories used in the analysis of releases of radioactive material are shown in Table 7. 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. For operating modes relating to refuelling outages, the stage of refuelling (i.e. before or 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.

10.21. The efficiency with which the groups of radionuclides listed in Table 7 are transported to the environment depends strongly on the chemical form they assume after leaving the core region. Numerous chemical interactions can occur that cause elemental forms of these species to react and form compounds with a wide range of physical properties [56]. 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 7 can be transported in more than one chemical form. Partitioning of the core inventory of reactive species among their possible chemical forms is considered a best practice that might introduce a source of uncertainty in the source

TABLE 7. 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	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	Cs
Alkaline earths	Ba, Mg, Ca, Sr, Be, Ra	Ba
Chalcogens	Te, O, S, Se, Po	Te
Refractory metals and platinoids	Ru, Mo[a], Pd, Tc, Rh, Re, Os, Ir, Pt, Au, Ni	Ru
Tetravalents	Ce, Pu, Np, Zr, U[a], Ti, 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.

term calculations. If performed, the impact of the partitioning in the assessment of source term calculations should be considered via a sensitivity study, for example.

Source term analysis with a fast-running code

10.22. 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 material 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 category attributes and are input data for the fast-running code.

10.23. For fast-running codes the grouping of radionuclides is defined in a similar way as for integral codes (see Table 7).

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

10.25. Uncertainties in the key parameters related to the accident phenomena mentioned in para. 10.22 should be considered in the fast-running code to allow for the quantification of uncertainties in the source term analysis.

10.26. A fast-running code may be integrated into the accident progression event tree so that the tree quantification includes the source term analysis including uncertainties.

Source term analysis with a simplified approach

10.27. In some cases, a simplified approach can be taken, using the source term analysis from another nuclear power plant whose design and features relevant to severe accident progression are sufficiently similar to the plant being analysed, and for which the results of deterministic analysis are available. When a reference source term analysis is used as a surrogate for plant specific calculations, the following criteria should be met for its use in a Level 2 PSA:

(a) A technical basis should be established to justify that the plant undergoing Level 2 PSA 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 and systems should be identified and compared.

(b) It should be ensured that the accident sequence(s) modelled in the reference source term analysis are sufficiently similar to the accident sequences of interest to the Level 2 PSA for the plant under analysis. Differences in the operation of reactor safety systems or containment associated systems can invalidate the applicability of a reference plant calculation to a particular PDS.[28]
(c) The calculation for the reference plant 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 open literature (e.g. scientific and technical publications).

10.28. The 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.

USE OF COMPUTER CODES FOR SOURCE TERM ANALYSIS

10.29. Models and correlations introduced in the computer codes that are used for source term analysis (i.e. integral or fast-running codes) are required to be verified and validated (see Requirement 18 of GSR Part 4 (Rev. 1) [2]).

10.30. 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. Recommendations on the selection of software, approaches and methods are provided in paras 3.16–3.18.

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[28] For example, many calculations of accident sequences involving ‘station blackout’ for several reactor designs can be found in 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.

RESULTS OF THE SOURCE TERM ANALYSIS

10.31. 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 8 (another example is presented in Ref. [57]).

10.32. The source terms and frequencies of the release categories — the latter 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. To achieve this, the terms 'large' and 'early' need 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. [57].

10.33. 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. To achieve this, the term 'quantity of release' needs to be defined within the Level 2 PSA project; this term 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.

10.34. 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).

TABLE 8. EXAMPLE SUMMARY OF SOURCE TERMS FOR WATER COOLED REACTORS

	Release category attributes							Fraction of core inventory[a] to environment[b]			
	Release category	Frequency (a^{-1})	Time release begins	RCS[c] 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[d]	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[e]	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] Fraction of core inventory to the environment here refers to the core inventory before the severe accident scenario begins.
[b] These are sample values only.
[c] RCS: reactor coolant system.
[d] SGTR: steam generator tube rupture.
[e] The ‘nominal leakage’ value is based on normal operating conditions (measured by tests), which might be different from design.

ANALYSIS OF UNCERTAINTIES IN SOURCE TERMS

10.35. 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 the calculated results. In addition to the uncertainties in modelling severe accident phenomena, many chemical and physical processes are still subject to research. These processes govern the release of radioactive material from fuel, the deposition and retention of radioactive material on reactor internal surfaces and containment surfaces, and radioactive material from scrubbing by containment associated systems. The following are examples of issues that give rise to uncertainties in source term analysis for water cooled reactors:[29]

(a) Uncertainties in core damage processes and containment behaviour (see Sections 6 and 7);
(b) Effects of fuel exposure (burnup) on the release fraction of radioactive material from the fuel matrix;
(c) Chemical forms of volatile and semi-volatile species;
(d) Chemical interactions with fuel, neutron absorbers and structural materials during core degradation;
(e) Deposition rates of radioactive material and aerosols on the surfaces of the reactor coolant circuit;
(f) Deposition of radioactive material in piping and other components in accident sequences with containment bypass;
(g) Release of radioactive material and aerosols during molten core–concrete interaction;
(h) Chemical processes in reactor coolant systems, corium and during molten core–concrete interaction;
(i) Interaction between hydrogen burn or radicals in flame fronts and airborne radioactive material (e.g. possible resuspension of radioactive deposits);
(j) Scrubbing efficiency of aerosols and vapours in suppression pools, ice beds or bubble towers;
(k) Aqueous chemistry of radioactive material captured in water pools;
(l) Revaporization and resuspension of radioactive material from surfaces;
(m) Chemical decomposition of radioactive material aerosols;

[29] Many of the examples given relate to iodine (and ruthenium) forms and behaviour. Owing to the importance of assessing radiological consequences, reduction of uncertainties is an issue to be addressed in dedicated severe accident research programmes.

(n) Radioactive release into the environment with regard to containment break size, containment leak rate, released fraction of inventory, iodine chemistry and deposition of radioactive material onto containment surfaces.

10.36. Past and ongoing research programmes have made significant progress towards reducing uncertainty in severe accident source terms (see, e.g., Refs [54, 55]). 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 the response of the containment to conditions beyond its design basis 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.

10.37. Up to date knowledge on severe accidents and on fission product behaviour should be used in Level 2 PSA. 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 11.25 and 11.26). Uncertainty modelling can also be introduced directly into the accident progression event tree (i.e. distribution of probability) for the propagation of uncertainties within the model while it is possible, depending on the PSA tool.

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 the 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: the first based on minimal cutset calculations (Boolean algorithm) and the second based on scenario calculations (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 [36]), 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 PSA, 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 performed using a suitable computer code that has been through a comprehensive process of verification and validation. Most PSA computer codes used for the Level 1 PSA are suitable for Level 2 PSA analysis, as well (i.e. computer codes used for the development of event trees and fault trees). To perform these analyses, the users of the codes should be adequately experienced in PSA modelling, should have a detailed understanding of the severe accident progression process and should understand the limitations of the code.

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

(a) Release category frequency;
(b) Contributions to the release category 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 (e.g. 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. To verify the correctness of the severe accident progression sequence modelling, the sum of the release category frequencies should be validated against the core damage frequency determined from the Level 1 PSA (typically core and/or fuel damage frequency). Justification for any numerical deviations should be given.

11.7. The analyst should check that the accident sequences or cutsets identified by the solution of the Level 1 PSA model are propagated into the Level 2 PSA 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 included in the list of cutsets (or sequences) generated. The software used for accident progression event tree quantification should be capable of quantifying success branches in the event tree, since success branches are equally important as failed branches in the accident progression event tree. In fact, the conventional meanings of success and failure do not always apply in accident progression event trees: 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 paras 2.16–2.19), release metrics associated with the containment failure and source term 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 of the Level 2 PSA cutsets (i.e. 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, post-processing of cutsets might 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 possible inconsistencies and should check whether the Level 2 PSA conserves the frequency input from the Level 1 PSA.

11.10. For quantification of the Level 2 PSA, truncation limits (e.g. cut-offs) should 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 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. Performing a study is a typical way to demonstrate convergence. In a convergence study, the analyst performs 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. The frequency of each release category should be indicated first.

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 containment performance matrix (‘C matrix’) [58], 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 and the conditional probabilities that a release category can be realized, given a specific PDS (see Eq. (1)). An example layout for the content of a C matrix is presented in Table 9. Uncertainty analysis leads to alternative sets of values of the elements of the C matrix:

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

where

R(n)	is the release frequency;
M	is the total number of PDSs;
F(m)	is the PDS frequency;
C(m, n)	is the conditional probability that a release category *n* can be realized, given a PDS *m*.

11.13. For each PDS, the main minimal cutsets could also be indicated. A variant of the C matrix is to identify the conditional probability that a release category can be realized, given the families of PDSs for the list of initiating events (see Eq. (2)). This might not be sufficient, so additional useful information may be presented, such as the release category frequencies for each plant operational state, or the distribution of the different causes of containment failure for specific release categories.

TABLE 9. CONTAINMENT 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

$$R(n) = \sum_{m=1}^{M} \sum_{j=1}^{J} A(j)\ B(j,\ m)\ C(m,\ n) \tag{2}$$

where

R(n)	is the release frequency;
M	is the total number of PDSs;
J	is the total number of initiating events;
C(m, n)	is the conditional probability that a release category *n* can be realized, given the families of PDSs, *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, but 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, the frequencies and uncertainties associated with each release category can be determined 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 contribution of each release category to the total release frequency should also be tabulated to allow for 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 should be 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 (see Section 10 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 take into account 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 accident sequence that contributes to a particular release category. An intermediate approach is sometimes taken, in which 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 analysis

11.18. Importance measures for basic events, groups of basic events, credited systems and groups of initiating events, among others, 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, but other potential end states may be of interest. These metrics may be more specific or may encompass more than one operating mode or operational state. Importance measures typically include: (a) Fussell–Vesely importance; (b) risk reduction worth; (c) risk achievement worth and (d) Birnbaum importance. The various importance measures provide a perspective on which basic events and other systems and events 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.

Types of uncertainty

11.20. 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 treatment of uncertainties is one of the most important aspects of Level 2 PSA.

11.21. Paragraphs 11.22–11.25 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 para. 10.35). Recommendations on uncertainties related to systems and operator actions are provided in SSG-3 (Rev. 1) [4].

11.22. Uncertainty arises in a Level 2 PSA analysis as a result of several factors. The analyst should consider the following sources of uncertainty or should develop, use and justify an alternative source, 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 consisting of cutsets, correctness of the input parameters and assumption of human errors. 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 and reviewed (see paras 3.23–3.29).

(b) *Loss of detail due to aggregation.* Loss of detail can occur at several points in the Level 2 PSA. 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 grouping (binning; see para. 1.9) accident sequences on the basis of key sequence attributes introduces bias and uncertainty, as the attributes used by the analyst to group accident sequences need to be broad. In Level 2 PSA, these grouped sequences can vary in the timing of the initial fuel damage, the impact of the event progression on the containment, and the magnitude and content of the containment fission product release, among other things. The grouping process often has two competing goals: optimizing the number of detailed severe accident progression analyses to be performed, and providing a sufficient level of detail for Level 2 PSA insights or for the characterization of the Level 3 PSA inputs. The impact of the grouping process on uncertainty is difficult or impossible to quantify precisely; however, this grouping process is intended to maintain the key features of the release category such that more refined analysis or subdivision will not impact Level 2 PSA insights. In practice, the grouping process is intended to be performed conservatively. As more detailed resolution regarding severe accident phenomenology becomes available, and as increases in computing resources allow increasing levels of detail to be captured in the PSA, this uncertainty is expected to diminish. Sensitivity analyses may be used to assess the extent of these modelling uncertainties on the plant response. To mitigate these uncertainties, the grouping within a Level 2 PSA should be justified and well documented.

(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 7.23–7.29), human and equipment reliability analysis (see paras 8.19–8.23), development of accident progression event trees (see paras 9.7–9.11) and source term analysis (see paras 10.35–10.37).

(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 of the parameters and performing sensitivity studies or propagating them through the analysis. Recommendations on addressing parameter uncertainties within Level 2 PSA are provided in paras 11.24–11.26.

11.23. 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 of the effects of these uncertainties is typically supported using two methods: uncertainty analysis and sensitivity analysis.

Uncertainty analysis

11.24. 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 5th, 50th and 95th 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 that the analyst should take, as follows:

(a) *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 [33] provides information on an evaluation of uncertainties in relation to severe accidents and Level 2 PSA.

(b) *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 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 should be considered for peer review as part of the independent verification (see paras 3.24–3.29).

(c) *Propagation of uncertainties*. Propagation of uncertainties is usually done by simulation methods based on either simple (Monte Carlo) random sampling or stratified (Latin hypercube) sampling procedures. Additional details can be found in Refs [25, 26, 59–66].

(d) *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 [25, 26]. Regression analysis techniques can also be applied to assess the importance of particular uncertain issues in the PSA (see, e.g., Ref. [33]). Correlation coefficients of dependent variables with respect to uncertain issues or phenomena can provide insights into their importance.

Sensitivity analysis

11.25. 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). Sensitivity analysis, specific to parameters, events and/or phenomena 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.

11.26. 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 LEVEL 2 PSA: PRESENTATION AND INTERPRETATION OF RESULTS

OBJECTIVES AND CONTENT OF DOCUMENTATION

12.1. Requirement 20 of GSR Part 4 (Rev. 1) [2] states that "**The results and findings of the safety assessment shall be documented.**" General recommendations related to the documentation of PSA results are provided in paras 3.17–3.19 of SSG-3 (Rev. 1) [4] and are not repeated here.

12.2. 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 report or as part of internal reports.

12.3. 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.

12.4. To support maintaining a living PSA, in line with Requirements 12 and 24 of GSR Part 4 (Rev. 1) [2] (see para. 2.20), 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 progression analysis, integration of new topics, use of improved models, broadening of the scope of the PSA in question and its use for alternative applications.

12.5. The documentation of the Level 2 PSA should explicitly present 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.

12.6. 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 and for facilitating the independent review of the Level 2 PSA.

12.7. Conclusions should be distinct and should not only reflect 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.

12.8. 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 made and should be presented with the final Level 2 PSA results.

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

(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 in particular for the containment;
(e) Significant accident scenarios contributing to a large release or any other applicable risk metrics (see paras 2.16–2.19).

12.10. 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 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 2.16–2.19 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.

12.11. 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 users (e.g. the public) might primarily use the summary report of the PSA, while others (e.g. regulatory bodies) 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 by the operating organization. This decision making process may include the PSA team and the project management team for the Level 2 PSA.

ORGANIZATION OF THE DOCUMENTATION

12.12. Recommendations on the organization and preparation of documentation for PSA are provided in paras 3.20–3.24 of 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 a 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.

12.13. 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 is generally aimed at a wide audience of reactor safety specialists and should be adequate for high level review.

12.14. 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 provide a comprehensive overview of the entire Level 2 PSA study. It should be independently reviewed by individual task leaders and/or analysts for correctness and consistency.

12.15. 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 Level 2 PSA should also be included in this subsection of the summary report.

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

(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 the 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, the uncertainties, and the sensitivity and importance analyses.

12.17. The main report, together with its appendices, should be designed to do the following:

(a) Support technical review of the Level 2 PSA;
(b) Communicate key detailed information to interested users;
(c) Permit the efficient and varied application of the Level 2 PSA models and results;
(d) Facilitate the updating of the models, data and results in order to support the continued safety management of the plant.

12.18. 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 Level 2 PSA report, as far as possible.

12.19. Reporting of the rationale and analyses employed for a Level 2 PSA should include information on the methods used, the PSA process, and the insights and conclusions drawn, presented 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.

12.20. Sample outlines of the contents of the summary report and main report for a Level 2 PSA are provided in Annex II.

COMMUNICATION OF RESULTS

12.21. Paragraph 5.9 of GSR Part 4 (Rev. 1) [2] states:

> "Consideration shall also be given to ways in which results and insights from the safety assessment may best be communicated to a wide range of interested parties, including the designers, the operating organization, the regulatory body and other professionals. Communication of the results from the safety assessment to interested parties shall be commensurate with the possible radiation risks arising from the facility or activity and the complexity of the models and tools used."

12.22. For communication of the Level 2 PSA results, the target audience and the amount of information and data provided should be taken into consideration. The various reports listed in para. 12.12 are suitable sources of information that could be communicated, depending on the target audience.

12.23. As reports on Level 2 PSA might contain sensitive information, security considerations should be taken into account for the communication of such reports. Guidance on security aspects related to information classification, sharing and disclosure is provided in IAEA Nuclear Security Series No. 23-G, Security of Nuclear Information [67].

13. LEVEL 2 PSA FOR A SPENT FUEL POOL

13.1. Interest in the risk related to accidents occurring in the SFP increased after the accident at the Fukushima Daiichi nuclear power plant [68, 69]. A Level 2 PSA for the SFP might not be necessary, given the generally low frequency, the long time frames for proceeding to fuel damage and the potential limited mitigation capabilities once fuel damage has occurred in the SFP. Depending on the pool's location and plant design specifics, available capabilities to prevent off-site releases in case of fuel damage at a SFP 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. The recommendations provided in this section are focused on the development of Level 2 PSA when the SFP is located inside a building capable of ensuring the confinement function in severe accident conditions (see para. 6.20). If this is not the case, the practice during Level 2 PSA has been to consider that accidents involving damage of fuel stored in the SFP lead directly to large radioactive releases. In addition, an analysis can be performed with the objective of 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 SFP is based on the same methodology as the Level 2 PSA for the reactor outlined in Sections 5–11. Accordingly, the general process for conducting a Level 2 PSA for the reactor can be adapted for the SFP, with additional consideration of the specific aspects addressed in this section.

13.4. The goal of performing a Level 2 PSA for the SFP should be clearly defined; usually this goal is similar to the goal of a Level 2 PSA for the reactor. A Level 2 PSA for the SFP can be performed independently or in combination with a Level 2 PSA for the reactor, 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 SFP (e.g. inside the reactor containment, outside the reactor containment but inside the reactor building, outside the reactor building) can determine the analysis needs for a Level 2 PSA for the SFP. For example, the location of the SFP determines whether an accident progression event tree needs be developed or whether other factors that could reduce the source term could be taken into consideration. The source term could be reduced by the possibility of closing the reactor containment and the availability of the ventilation system and SFP cooling system.

INTERFACE WITH LEVEL 1 PSA FOR A SPENT FUEL POOL

13.5. As when performing a Level 2 PSA for a reactor, PDSs specific to the SFP can be considered in the development of a Level 2 PSA for the SFP. Factors specific to SFP analysis include items that influence the accident progression and source term, such as: time since last core unloading, the fuel loading in the pool (e.g. number of fuel assemblies, fuel burnup, fuel loading pattern), pool configuration (i.e. whether the SFP is isolated from or interconnected to the reactor or to the SFPs 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 or handled in the SFP, boiling of the pool water) defined in Level 1 PSA for the SFP, as described in paras 10.2–10.6 of SSG-3 (Rev. 1) [4], should also be addressed in the Level 2 PSA.

13.7. If the Level 2 PSA for the SFP and the Level 2 PSA for the reactor are integrated in the same model, the PDSs should include the status of both the reactor and the SFP for events that can affect both. Reactor accident sequences might impact the SFP; for example, reactor containment venting might accelerate the boiling of the water in the SFP if the SFP is located inside the reactor containment. In addition, reactor accident sequences that do not result in Level 1 reactor core damage events might impact the mitigatory actions for the SFP accidents and therefore might need to be considered for inclusion in the PDSs.

SEVERE ACCIDENT PROGRESSION ANALYSIS OF FUEL STORED IN THE SPENT FUEL POOL

13.8. To support Level 2 PSA for an SFP (if necessary; see para. 13.2), deterministic analyses should be performed to analyse severe accident progression in the SFP using one or more computer codes capable of modelling the accident progression and severe accident phenomena in the SFP. Severe accident phenomena to be considered in this analysis include heat transfer within the pool and to the fuel racks and surrounding walls (and to the ground, for SFPs below ground level), fuel and cladding behaviour (e.g. fuel burnup, decay heat, cladding behaviour), fuel assembly and rack degradation including interactions (e.g. zirconium clad and water reaction, hydrogen generation, zirconium fire, corium–concrete interaction), and fission product transport. The calculations made should provide information on the fraction of the fuel assemblies that would be damaged, depending on the fuel assemblies’ arrangement, burnup and storage time in the SFP.

13.9. The boundary conditions should be defined in accordance with the PDS. In particular, care should be taken in defining in the calculations the amount of fuel that is normally replaced during a refuelling outage and during a full core unload (if a full core unload is prescribed by the operating procedures and is within the scope of the plant operational states included in the PSA).

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

13.11. The following accident progression parameters should be considered for the SFP:

(a) Time to boiling;
(b) Time to fuel uncovering;
(c) Time to fuel damage;
(d) Time to spent fuel structure breach;
(e) Time to penetration of concrete around the SFP (if credited);
(f) Source term magnitude and timing.

13.12. Depending on the plant configuration (e.g. SFP inside the reactor containment, SFP outside the reactor containment but inside the reactor building, SFP outside the reactor building), severe accident progression analysis should consider the interactions between the reactor and the SFP, as a reactor accident might have an impact on or induce an SFP accident and vice versa. From this analysis, some additional accident scenarios (involving both the reactor and the SFP) could be incorporated into the Level 2 PSA if not already considered in the Level 1 PSA, such as the following:

(a) Common events that have an impact on the reactor and the SFP simultaneously (e.g. station blackout) or consequential failure during accident progression, which affects the safety functions for the other source;
(b) Impact of the accident management strategies for the reactor on the SFP (e.g. if the SFP is located inside the reactor containment, actuation of the filtered containment venting system leads to more intensive water boiling in the SFP);
(c) Hydrogen release that could result in deflagration or detonation events that lead to the failure of structures or electrical and/or 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 SFP on the containment structure or steel liner;
(f) Subsequent independent failure of installed equipment that ensures safety functions for both the reactor and the SFP (e.g. residual heat removal), which might force the operators to decide where to prioritize the use of any remaining non-permanent equipment.

13.13. If the SFP is located inside the reactor containment, accident progression analysis should address the impact of a combined reactor and SFP accident on conditions inside the reactor containment (e.g. pressure, temperature, corium spreading, inflammable gas).

13.14. Hydrogen generation should be considered for loss of heat removal scenarios that cause the evaporation of a large amount of water from the SFP. For SFPs located inside 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 to ensure that they are sufficient to cope with the hydrogen additionally generated from the damaged fuel stored in the SFP.

13.15. Generated hydrogen might leak into rooms adjacent or connected to the SFP. The retention of hydrogen in the rooms on the release path, which might lead to combustion and damage to the rooms, should be justified.

13.16. For fuel utilizing zirconium as fuel cladding, the risk of zirconium fire and its propagation under dry conditions in the SFP should be considered.

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

13.18. For an SFP located inside a building capable of ensuring the confinement function, the overpressure created by steam and heat 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. The consequences of accidents during fuel transfer operations between the reactor and the SFP, if not already screened out, should be considered for dedicated analysis in the Level 2 PSA. Typical accidents to be considered are related to fuel uncovering due to the loss of the spent fuel cooling system (e.g. as a result of a station blackout or an external hazard such as a seismic event).

ACCIDENT PROGRESSION EVENT TREE FOR A SPENT FUEL POOL

13.20. If the SFP is located inside the containment, the accident progression event tree can be developed similarly as for the reactor Level 2 PSA (see Section 9). Nevertheless, the analyst should introduce the dependencies between the reactor and the SFP (e.g. systems, human actions, containment response) in the accident progression event tree. In comparison with a stand-alone Level 1 PSA for an SFP, some additional scenarios of SFP accidents induced by a reactor core melt accident may be introduced. For an SFP outside the containment, the accident progression event trees for the Level 2 PSA can be simple event trees, depending on the extent to which mitigation strategies are credited.

13.21. The harsh environments (e.g. high temperature, humidity and radiation levels) expected to be present around the SFP should be carefully considered when crediting local operator actions in the Level 2 PSA. The effect of these harsh environments on the survivability of mitigative equipment such as hoses, on the fittings and nozzles present around the SFP and on the availability of the instrumentation system (e.g. SFP water level) should also be considered in the Level 2 PSA.

SOURCE TERM AND RELEASE CATEGORIES FOR A SPENT FUEL POOL

13.22. The source term calculations for an SFP Level 2 PSA can be performed similarly to those performed for the reactor Level 2 PSA (see Section 10).

13.23. Release paths from the SFP to the environment that are to be considered in the Level 2 PSA depend on the location of the pool. If the SFP is located inside the reactor containment, the potential release paths to the environment are almost the same as for reactor core melt accidents. However, additional release paths should also be reviewed (e.g. after penetrating the concrete wall or bottom of the SFP, the molten debris could penetrate the containment wall, leading to another type of late containment failure). If the SFP is located outside the reactor containment, the potential release paths to the environment depend on plant specific designs, such as ventilation systems, building doors, roof under thermal impact, and size of rooms on the path.

13.24. In the case of a severe accident in the reactor concurrent with significant fuel damage in the SFP, the source term evaluation should consider the different timing of radioactive releases from those two sources.

13.25. The deposition of aerosols in the buildings in the release paths could mitigate the environmental impact and should be considered.

13.26. A dedicated source term analysis should be performed for the SFP and for accidents involving the transfer of fuel assemblies between the reactor and the SFP on the basis of the burnup of the fuel elements. The fuel inventory of the SFP at potential accident times should be analysed considering the history of refuelling and the subsequent mixture of newer and older fuel elements.

13.27. Similarly to in the reactor Level 2 PSA, release categories can be defined for the SFP in order to group similar accident sequences based on the magnitude and timing of the release and to calculate an associated frequency of release.

QUANTIFICATION OF EVENT TREES AND ANALYSIS OF RESULTS FOR A SPENT FUEL POOL

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

14. LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.1. Paragraphs 14.2–14.29 provide recommendations on the development of Level 2 PSA for sites where multiple units are located, suitable for use where national regulatory requirements compel such studies. Given the complexity of models and the high level of uncertainty, the development of such Level 2 PSAs is not yet common practice among States. However, it can be useful to capture risks relevant to the whole site as well as dependencies among units from the Level 2 PSA perspective if these risks and dependencies 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 States. More information on Member States' experience, practical case studies and guidance on PSA for multi-unit nuclear power plants are provided in Refs [70, 71].

OBJECTIVE OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.2. The objective of the development of Level 2 PSA for multi-unit nuclear power plants is to complement the Level 2 PSAs for each unit on the site, with regard to topics that might have not been fully addressed, such as equipment and systems that are shared or common to multiple units.

SCOPE OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.3. In developing the Level 2 PSA for multi-unit nuclear power plants, the analyst should identify a list of topics that have not been fully addressed in the single unit Level 2 PSA, such as the following:

(a) Hazards affecting the whole site, especially the implications for the severe accident mitigation system, equipment reliability and for human resources;
(b) Correlated or shared SSCs and resources among different units;
(c) Plant operational states of each unit on the same site (see Section 5 for considerations of operational states other than full power);
(d) Impact on the other units of consequences induced by a unit with a severe accident (e.g. fuel melt accidents happening in another unit).

The selection of topics should not lead to excessive complexity in the development of the Level 2 PSA for multi-unit nuclear power plants.

14.4. The analyst should justify whether or not those topics are introduced in the multi-unit Level 2 PSA on the basis of a screening process. A Level 2 PSA for a multi-unit nuclear power plant should consider the reactor units (e.g. power and/or research reactors) and the SFPs on the site. When a single reactor unit includes multiple reactor modules, the Level 2 PSA for that reactor unit should address interactions and dependencies among the reactor modules in the same way that reactor units are addressed in the Level 2 PSA for a multi-unit nuclear power plant. For the purpose of this Safety Guide, reactor modules are understood as nuclear reactors that are in the same reactor unit, in accordance with Ref. [72].

PREREQUISITES OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.5. The recommendations provided in paras 4.11–4.17 related to plant familiarization for the development of a single unit Level 2 PSA are also applicable for the development of the Level 2 PSA for multi-unit nuclear power plants.

14.6. The availability of a multi-unit Level 1 PSA model for a multi-unit site is a prerequisite for the development of Level 2 PSA for a multi-unit site.

RISK METRICS FOR LEVEL 2 PSA FOR A MULTI-UNIT NUCLEAR POWER PLANT

14.7. The risk metrics used in PSA for a single unit site (e.g. large release frequency) should be used to the greatest extent possible to express the risk profile in the context of multi-unit nuclear power plants for corresponding decision making (see paras 2.17–2.19). These risk metrics could be adapted to specific multi-unit risk metrics, for example the conditional probability of large releases from several reactors, on the basis of the large releases from one reactor of a unit on a multi-unit site.

INTERFACE BETWEEN LEVEL 1 PSA AND LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.8. The Level 2 PSA for a multi-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 into the Level 2 PSA.

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

14.10. PDSs for accidents involving multiple units should group accident sequences coming from a Level 1 PSA for multi-unit nuclear power plants that are equivalent in terms of the risk of release considering the parallel evolution of all units on site (e.g. kinetics of accident progression, availability of mitigating systems).

ACCIDENT PROGRESSION AND CONTAINMENT ANALYSIS IN LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

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

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

14.13. The major cause of differences 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 6–8. 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 MULTI-UNIT NUCLEAR POWER PLANTS

14.14. The recommendations provided in Section 8 should be considered applicable for Level 2 PSA for multi-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 multi-unit assessment. Specific aspects of Level 2 PSA for multiple units (e.g. recovery post-core damage, ensuring containment performance) necessitate the consideration of the state of other units on the site in terms of available resources and feasibility of specific Level 2 actions.

14.15. For plants with multiple units, further to the interactions between the units (both positive and negative, from a risk point of view) considered in Level 1 PSA from the perspective of the unit under consideration, the shared systems between units and the degree of shared responsibility in the operator response to accidents should be considered in the Level 2 PSA.

14.16. The dominant consideration for human reliability analysis purposes is 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, the ability of those responses to occur in parallel on multiple units should be considered.

14.17. Sites with multiple units of the same design are more likely to have 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, however. In the event of concurrent events at multiple units, the impact on operator actions might have implications for Level 1 PSA for multi-unit sites that should be considered for Level 2 PSA for multi-unit sites.

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

14.18. The development of an accident progression event tree for Level 2 PSA for multi-unit nuclear power plants should be based on the recommendations provided for single unit plants in Section 9.

14.19. The degree of interconnectedness between units should be taken into account. Consideration should also be given to the sharing of SSCs between units as well as the potential consequences of one unit on the undisturbed operation or mitigation of accident conditions on another unit.

14.20. Since the number of units can add significant complexity and size to the accident progression event tree, consideration should be given to simplifying the single unit models before combining them for multi-unit plants, with account taken of major risk contributors. Since each Level 1 sequence results in multiple Level 2 sequences by definition, it is prudent to simplify them where possible. Methods to simplify the modelling could include the justified removal of low risk initiating event contributors, a focus on 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) and 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]

SOURCE TERM AND RELEASE CATEGORIES IN LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.21. 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 multi-unit nuclear power plants, since they generally apply the same assessment of source terms and the resulting release category assignments. Much of the existing framework for release categories may be used as is, with consideration given to where interconnectedness and dependencies for multi-unit Level 2 PSA may arise (as captured in the interface with multi-unit Level 1 PSA).

14.22. In the analysis of source terms and release categories in Level 2 PSA for multi-unit nuclear power plants, account should be taken of small and early

[30] For example, for a three unit site, grouping the end states into those for one, two and three units resulting in releases, as opposed to modelling each possible combination of units.

releases from different sources at the site, as these releases might become relevant when aggregated for the calculation of the large release or large early release risk metrics for the site.

14.23. 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 multi-unit Level 2 PSA.

14.24. It may be justifiable to further reduce the total number of release categories for multi-unit Level 2 PSA by considering 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 an additional level of detail is needed for the purposes of Level 2 PSA for multi-unit nuclear power plants in comparison to a single unit analysis.

14.25. The addition of SFP accidents or shutdown states to the scope of a Level 2 PSA for multi-unit nuclear power plants might also further complicate the source term analysis for multi-unit PSA due to the further increase in combinations of outcomes (even beyond those discussed in para. 13.28). By similar reasoning, screening and simplification of those outage states and radioactive sources can be applied.

14.26. The aggregation and simplified combination of release categories for various nuclear installations on a given site can be further assessed on the basis of the commonality or differences in the definitions of releases from various nuclear installations. While a reactor and SFP may have differing characteristics of core or fuel damage, it is easier to define releases from either nuclear installation as exceeding a defined release magnitude and facilitate easier grouping (binning) of contributors to a single release category.

QUANTIFICATION OF EVENT TREES AND ANALYSIS OF RESULTS IN LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.27. The integration and quantification process for Level 2 PSA for multi-unit nuclear power plants should be based on the approach used in the single unit Level 2 PSA (see Section 11). In the case of coupling PSA models from different units into a single PSA model, the major concern is the additional complexity resulting from the additional event tree end states, release categories and combinations. Quantification is expected to involve additional consolidation and screening to include a manageable set of inputs for Level 2 scenarios that take into account the effect of multiple units undergoing Level 1 and Level 2 aspects. Depending on the topics of interest identified (see para. 14.3), an approach based on the post-processing of the single unit Level 2 PSA results could be sufficient to obtain relevant insights.

14.28. The treatment of sensitivity and uncertainties does not bear any major methodological differences to Level 2 PSA for single unit nuclear power plants (see paras 11.24–11.26), but it should be expected that a manageable addition for specific multi-unit impacts via sensitivity cases for phenomenology and modelling uncertainties will be necessary to appropriately characterize the state of knowledge for multi-unit Level 2 PSA.

DOCUMENTATION OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.29. The recommendations provided in Section 12 should be considered applicable to a Level 2 PSA for multi-unit nuclear power plants, with no additional areas that need to be addressed. However, additional details and discussion regarding the characteristics of releases, the phenomenology that might arise due to accidents involving multiple units at a nuclear power plant, 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 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]), often in combination with Level 2 PSA results. The following list includes some examples of applications of Level 2 PSA (these applications 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) Development of a list of severe accident scenarios to be addressed in the nuclear power plant design.

SCOPE AND LEVEL OF DETAIL OF LEVEL 2 PSA FOR VARIOUS USES AND APPLICATIONS

15.2. The scope and the level of detail of the Level 2 PSA should be consistent with its intended uses or applications. For example, the scope and the level of detail of a Level 2 PSA intended to provide an estimate of the large release frequency or the large early release frequency and to provide insights into the potential failure modes of the containment is different from the scope of a Level 2 PSA intended to provide an input to 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, to a limited extent, the frequencies associated with the occurrence of such releases, need to be specified more accurately. For Level 3 PSA development, both the source terms and their frequencies need to be specified more accurately. In addition, the level of detail of the PSA needs to be greater if the Level 2 PSA model is intended for use in a risk monitor. Recommendations on the risk monitor application in relation to PSA are presented in SSG-3 (Rev. 1) [4]. These recommendations are valid for the SSCs added in the Level 2 PSA model and are therefore not repeated in this Safety Guide.

15.3. The scope of the Level 2 PSA should be commensurate with its intended uses and applications (see also para. 2.7) and is based on the scope of the Level 1 PSA. A full scope Level 2 PSA is most suitable for a large number of uses and applications, with due consideration given to the uncertainties in key parameters and limited strength of knowledge on some data and assumptions that could impact the PSA results and insights. A full scope Level 2 PSA can be developed only on the basis of a full scope Level 1 PSA, as defined in para. 2.2 of SSG-3 (Rev. 1) [4].

15.4. If risk insights are to be derived from a Level 2 PSA that is not full scope (e.g. not all initiating events and hazards considered), this should be recognized when applying the insights from the PSA.

15.5. A full scope PSA ensures that the insights from the PSA relating to the risk significance of accident sequences, SSCs, human errors and common cause failures are derived from a comprehensive, integrated model of the nuclear power 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 NUCLEAR POWER PLANT

15.6. Level 2 PSAs should be actively maintained and periodically updated, taking into account changes in plant design and operational practices as well as feedback from experience and advances in technology that might compromise the validity of the PSA (see also paras 2.20–2.23). This updating should take account of changes in the provisions made and the guidance provided for severe accident management, updates to the severe accident progression analysis performed to support the Level 2 PSA model, and research results that provide a better understanding of the phenomena that occur during a severe accident.

15.7. The Level 2 PSA should 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 and should be updated throughout the construction and operational stages of the lifetime of the plant.

15.8. The Level 2 PSA should also 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 [73, 74].

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

(a) Any mandatory requirements that relate to the PSA application under consideration (e.g. legal requirements or regulations);
(b) The insights from deterministic safety analysis (e.g. whether the provisions of defence in depth requirements are being met; whether there are adequate safety margins; whether lower level requirements such as the provision of sufficient levels of redundancy and diversity in the SSCs that perform safety functions are being met; whether the equipment in the plant has been qualified to a sufficient level that it can withstand the harsh environmental conditions that follow initiating events);
(c) Any other applicable insights or information (e.g. a cost–benefit analysis, details of the remaining lifetime of the plant, inspection findings, operating experience, doses to workers that would arise if necessary changes were made to the plant hardware, environmental protection concerns).

COMPARISON OF LEVEL 2 PSA WITH PROBABILISTIC SAFETY GOALS OR CRITERIA

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 performed 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 will be exceeded.

15.13. Probabilistic criteria for a large off-site release of radioactive material needing a short term off-site response were proposed in Ref. [10].[31] Several States have also set similar numerical values, which have generally been defined as objectives or targets (see Annex III).

15.14. In addition, for future nuclear power plants, rather than defining probabilistic criteria, Ref. [10] states that the objective should be as follows:

> "[T]he practical elimination of accident sequences that could lead to large early radioactive releases, 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."

LEVEL 2 PSA FOR DESIGN EVALUATION

15.15. The Level 2 PSA should be used to perform 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 paras 12.19–12.46 of 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 the following:

(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 depends 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).

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

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 the following:

(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 safety features provided for the prevention or mitigation of severe accidents in order to reduce contributions to the overall risk of sequences with the highest risk significance. The improvements considered should include the provision of additional protective systems and features for mitigating the consequences of a severe accident. This could involve incorporating such additional protective systems and features into a new design or backfitting them into an existing plant.

15.19. 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 para. 2.13 of SSR-2/1 (Rev. 1) [3].

Comparison of design options

15.20. 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 in accordance with paras 2.24–2.36.

15.21. 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 comparison is done depends on the complexity of the design options being considered but could range from producing a revised PSA model to post-processing the cutsets to take account of simpler changes, or even to performing sensitivity studies that relate to the design options. In doing this comparison, it needs to be recognized that a design change might 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 should 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 might lead to combustible conditions in some time frames or even lead to concerns about containment underpressure.

USE OF LEVEL 2 PSA IN THE DEVELOPMENT OF SEVERE ACCIDENT MANAGEMENT GUIDELINES

15.22. The Level 2 PSA should be used as a basis for evaluating the measures in place and the actions that can be taken to mitigate the effects of a severe accident after core damage has occurred. The aim of such 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 taken for pressurized water reactors include the following:

(a) Opening the pressurizer relief valves in order to reduce the reactor coolant system pressure and thus 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.

15.23. 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, regardless of whether they have been specified using the Level 2 PSA or by another method.

15.24. In developing a Level 2 PSA, the uncertainties associated with the probability of phenomena occurring in the course of a severe accident should be recognized. The interrelations between these phenomena should also be noted, as an accident management measure aimed at mitigating a particular phenomenon might make another phenomenon more likely. Examples of such interrelations 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.

15.25. 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

15.26. 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, which leads to a significant level of uncertainty in the predictions of the Level 2 PSA.

15.27. The Level 2 PSA should provide a basis for the identification and prioritization of research activities. Such research activities should focus on identifying areas of research that can contribute most to improving knowledge in order to reduce the uncertainty in the highest risk significance parameters or phenomena.

INPUT FOR LEVEL 3 PSA

15.28. 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 a radioactive release 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.

15.29. 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 of the information needed for the Level 3 PSA is available in the Level 2 PSA end states.

15.30. The source terms and frequencies derived in the Level 2 PSA should be used as the starting point for the Level 3 PSA performed 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 dispersion of radioactive material and its release from the plant.

EMERGENCY PREPAREDNESS AND RESPONSE

15.31. The source terms and, to a limited extent, their frequencies derived in the Level 2 PSA, along with projections of the off-site dose as a function of distance, may 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.

15.32. 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 6).

15.33. Requirement 4 of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [75] 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 the use of the safety assessment as part of the integrated risk informed approach, the source terms and, to a limited extent, their frequencies derived in the Level 2 PSA can be used as an input to determine the emergency planning zones and emergency planning distances.

OTHER PSA APPLICATIONS

15.34. 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 provides 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 (e.g. large release frequency, large early release frequency) than for Level 1 PSA results (e.g. core damage frequency).

Appendix

CONSIDERATIONS FOR HUMAN RELIABILITY ANALYSIS IN A LEVEL 2 PSA

A.1. Significant differences in context exist for nuclear power plant operators performing tasks before and after core damage to protect the plant and the public. These differences significantly impact the factors to be considered when performing human reliability analysis for a Level 2 PSA. Example factors include command and control, coordination between the organizations involved in responding to the event, training of the nuclear power plant staff and the other organizations in responding to the events, specific procedural instructions, predictability of scenario progression, degraded plant instrumentation and the effect of the environment on task performance. A large amount of experience has been accumulated in performing human reliability analysis for Level 1 PSA, but much less for Level 2 PSA. This appendix discusses the differences in context and their implications in performing human analysis assessment in Level 2 PSA.

A.2. When core damage is imminent or occurring, the operator enters 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 radioactive material to the environment. In this new command and control structure, the technical support centre may decide or propose the mitigation strategies. Operators in the main control room implement the mitigation 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.

A.3. Multiple organizations may participate in responding to severe accidents. For example, the local fire brigade and companies with contracts with the plant for emergency response could come on to the site to support event mitigation. The government and police could evacuate residents. Ambulances could come onto the site to pick up injured workers. Organizations that have not participated in 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 issuing instructions for a controlled radioactive release if the line of command is not clear; as a result, the release could impact evacuation.

A.4. Most nuclear power plants routinely use plant specific, full scope plant simulators to train their operating personnel 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 a core crust shields the molten core from cooling water, the instrumentation might not provide information for the operator to evaluate the effectiveness of the cooling approach or information on whether the reactor vessel has remained intact. In general, severe accident mitigation involves the implementation of specific sensors, qualified for severe accident conditions, in order to improve the success of mitigatory actions. Because of the significant uncertainty in scenario progression, the severe accident management guidelines might be written such that operating personnel 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.

A.5. Environmental factors such as high temperatures, high levels of radiation, seismic aftershocks and blockage of routes for equipment transportation can affect task performance, as was the case 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 transport 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 availability and accuracy of instrumentation needed for operators to make decisions.

A.6. Paragraphs A.2–A.5 relate to factors in severe accidents that affect the cognitive functions of operating personnel in detecting plant information, understanding the plant status, making event mitigation decisions and implementing mitigation strategies. In addition, there are two significant differences between human reliability analysis practices in Level 2 PSA and Level 1 PSA. The first difference relates to the repair of unavailable components. Both emergency operating procedures and severe accident management guidelines may instruct operating personnel to repair unavailable components for event mitigation. While repair actions are hardly credited for Level 1 PSA, the longer time frame for Level 2 PSA makes repairing components an option. Operator interventions to mitigate severe accidents could still be needed even a few days after the core damage. The human reliability analysis community needs 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 to start the sequence with the PDS. Each PDS consists of many Level 1 scenarios. This practice simplifies the modelling of Level 2 PSA but it might have an influence on performing detailed human reliability analysis. If a scenario is started from a PDS instead of an initiating event, depending on the level of detail of the PDS defined, important scenario specific information might not be carried to Level 2 PSA for detailed human reliability analysis. The missing information, such as initiating events, failures of the SSCs, the availability of electric power and previous operator errors, may be important for dependency analysis. One way of reducing the effects on PSA results is to analyse the dominant minimal cutsets specific to the PDS to identify their representative event sequences from the initiating events to model dependency effects.

A.7. A few human reliability analysis 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 [76, 77], Human Action Modelling Standardized Tool for Editing and Recording (HAMSTER) [78], the Assessment Method for the Performance of Safety Operations (MERMOS) [79], Integrated Human Event Analysis for Event and Condition Assessments (IDHEAS-ECA) method [80], and expert judgment using a human reliability assessment process [81]. In addition, Ref. [82] provides guidance on applying the existing human reliability analysis methods to assess the human reliability of performing tasks in extreme conditions.

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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 that can be realistically examined using complex computer codes. This annex provides insights into the types of code typically used in Level 2 probabilistic safety assessments (PSAs) and a brief description of their areas of application. The most common codes are presented in Refs [I–1] to [I–32]. Severe accident phenomena encountered in a light water reactor are presented in Fig. I–1.

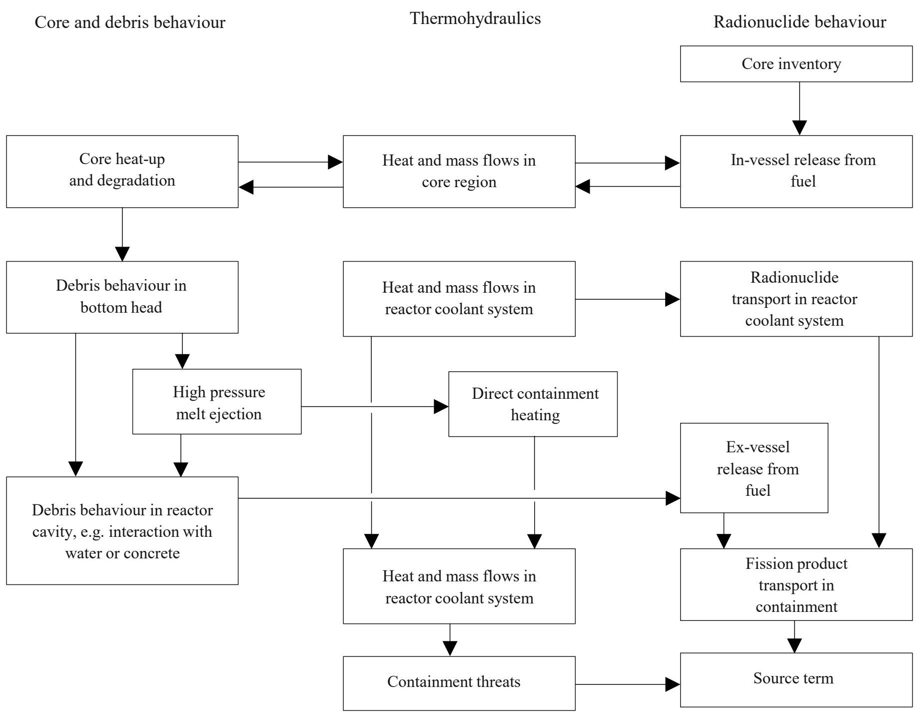

FIG. I–1. Severe accident phenomena encountered in light water reactors.

TYPES OF CODE USED FOR LEVEL 2 PSA

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, in accordance with their capabilities and intended use: mechanistic codes (see paras I–4 to I–5), integral codes (see paras I–6 to I–7) and dedicated codes (see paras I–8 to I–9).

I–3. In the past, separate codes, each dealing with a particular phase or aspect of severe accident behaviour, were coupled in a suite with an interfacing facility for the transfer of information between the codes. However, for routine PSA applications, it is desirable to have automatic transfer of information between the elements of a code suite, as manual transfer is slow and can 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 if a specific mechanistic model alone is used.

Mechanistic codes

I–4. Mechanistic codes (also referred to as 'detailed codes') calculate governing phenomena with best estimate models based on first principles, with computational resources being of secondary importance. Mechanistic codes are typically used in research to design and analyse severe accident experiments and to simulate, as accurately as possible, the phenomena and the behaviour of the nuclear power plant structures, systems and components during a severe accident. Once validated against appropriate experimental conditions and tests (i.e. integral or simplified tests), mechanistic codes 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. The main features of selected mechanistic codes are briefly described in Ref. [I–33].

I–5. Mechanistic codes generally provide an evaluation at a higher level of detail than needed 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.

Integral codes

I–6. Integral codes that are designed for routine application in PSA generally use simplified models of certain 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, integral 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 (e.g. fuel damage, melting) is simpler than the approaches used in mechanistic codes. Whereas in a mechanistic code, models might be used to evaluate explicitly the individual effects of several damage mechanisms within fuel rods, the same effects might be evaluated in a simpler and composite manner in integral codes.[1] 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–34, I–35]. The main features of selected integral codes are briefly described in Ref. [I–33].

I–7. 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. Examples of phenomena and processes modelled by such codes include the following:

(a) Thermohydraulic processes in the reactor coolant system, the containment structure and/or the structures, systems and components 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;

[1] For example, for cladding failures, mechanical codes might evaluate thermomechanical interactions between the swollen fuel pellet and bounding cladding, and local ballooning at weak points in the cladding, whereas such failures might be represented by specifying an effective cladding failure temperature in integral codes.

(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, carbon monoxide), transport and combustion;
(g) Radioactive material release (aerosol and vapour), transport and deposition;
(h) Behaviour of radioactive aerosols in the reactor containment, 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.

Dedicated codes

I–8. Dedicated codes and algorithms (also referred to as 'fast-running codes') provide rough estimates of parameters for specific PSA applications, such as estimation of the radiological source term [I–36] or of containment loads accompanying high pressure melt ejection [I–37]. Such codes and algorithms 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 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.

I–9. 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–31, I–32].

VALIDATION OF A CODE

I–10. The verification and validation of computer codes is crucial for enhancing confidence in their application. Achieving validation of severe accident codes is very difficult, as the extreme conditions that occur in a severe accident and the scale of the physical geometry are difficult to replicate in experiments. To conduct the process of validation, a validation matrix involving many simulations is generally needed. Caution needs to be applied if code validation has been performed 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 CODES

I–11. Deterministic accident analysis codes need to be designed so that a Level 2 PSA analyst familiar with general accident phenomena can run them reliably without needing to have the same detailed knowledge as a specialist using the code. In order for the code calculations to be meaningfully incorporated into the framework of a Level 2 PSA, the analyst needs to have a sound knowledge of the reactor systems and 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–12. The user needs to have a sound knowledge of the strengths and limitations of the code. Moreover, the code may not be used out of the range of situations and conditions for which it has been designed.

CODES DEVELOPED SPECIFICALLY FOR LEVEL 2 PSA

I–13. Codes for simulating fault trees and event trees and other simulation codes that are typically used for Level 1 PSAs are also often used for Level 2 PSAs. In many cases, such codes have been adapted or enhanced to address certain unique needs 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 separate Level 1 and Level 2 PSA modelling, there are several codes designed specifically for the modelling of Level 2 aspects (e.g. accident progression after core damage). Such codes have special features, such as user defined functions that calculate the severe accident event probabilities. 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 need to be adapted to model the behaviour of systems.

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Annex II

SAMPLE DOCUMENTATION FOR A LEVEL 2 PSA STUDY

II–1. Given the great number of uncertainties associated with the performance of Level 2 first use in Annex II, standardized contents of the reports presenting Level 2 probabilistic safety assessment (PSA), studies allow for a more effective peer review process. A sample outline of the contents of the summary report and the main report are presented below.

SAMPLE CONTENTS OF THE SUMMARY REPORT FOR A LEVEL 2 PSA STUDY

S1. Introduction

S2. Overview of the objectives and justification for the Level 2 PSA study

S3. Overview of the approach

S4. Containment failure modes and their 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. Outline of the main report

SAMPLE CONTENTS OF THE MAIN REPORT FOR A LEVEL 2 PSA STUDY

M1. Introduction

M1.1 Background

M1.2 Objectives

M1.3 Scope of the Level 2 PSA 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 with Level 1 PSA

M3.1 Grouping of accident sequences and specification of attributes

M3.2 Plant damage states for internal initiating events at full power and associated uncertainties

M3.3 Plant damage states for internal and external hazards at full power and associated uncertainties

M3.4 Plant damage states for other plant operational 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 hazards

M5. Accident progression and containment analysis

M5.1 Severe accident progression analysis

M5.1.1 Scope of analysis

M5.1.2 Method of analysis (e.g. codes, models)

M5.1.3 Summary of point estimate results for plant damage states 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 Grouping (binning) of accident progression event tree end states

M5.2.5 Treatment of uncertainties

M5.2.6 Results

M5.2.6.1 Point estimate containment performance matrix (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 radionuclides

M6.2 Method of analysis (e.g. codes, models)

M6.3 Summary of point estimate results for plant damage states 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

APPENDICES

A1. Basis for containment structural fragilities

A2. Basis for accident progression event tree quantification

A3. Results of deterministic severe accident progression 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 several factors such as the level of expertise, the preparation of the team, the scope of the Level 2 PSA, the regulatory process for review and approval, as well as resources involved. A Level 2 PSA takes from one year to several years to implement. Figure II–1 shows an example plan of Level 2 PSA related activities. The timing estimates are based on the assumption that the team performing the PSA is well trained.

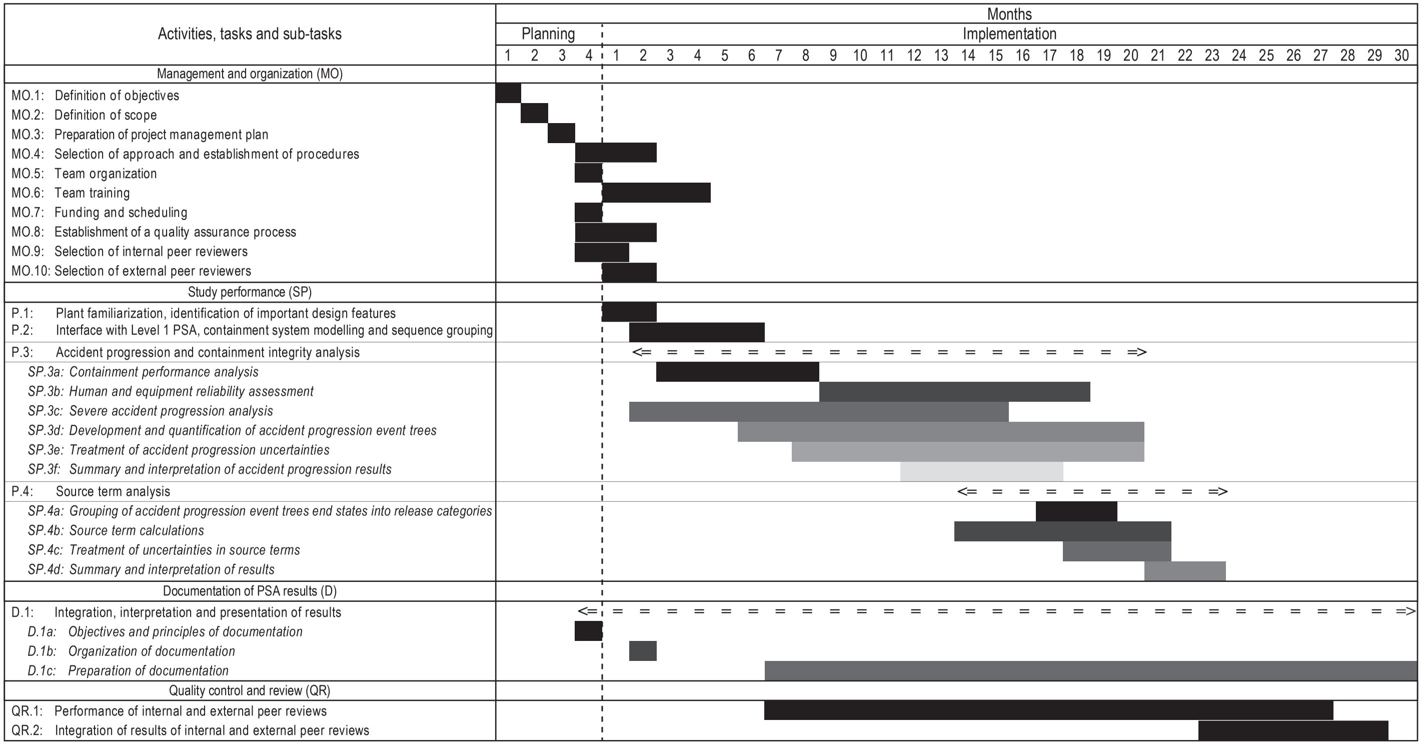

FIG. II-1: Example plan of activities for performance of a Level 2 PSA.

Annex III

EXAMPLES OF COMMON RISK METRICS IN LEVEL 2 PSA

III–1. Large release frequency and large early release frequency are the most common measures of risk used in Level 2 probabilistic safety assessment (PSA). In many 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 on the frequency of release, the main radionuclides in each release category and the notion of the time of the release. The 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 for large release frequency should provide information on both the frequency of release and the main radionuclides in each release category, integrated over a specified period of time.

III–2. Tables III–1 and III–2 provide examples of some Member States' values and definitions of large release frequency and large early release frequency, respectively. The examples are based on the original regulatory texts (and direct translation into English, as appropriate) and the regulatory framework for safety in each Member State is to be taken into account when viewing the tables.

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES

Member State	Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
Bulgaria	[III–1]	Large releases shall mean releases of radioactive material to the environment, which necessitate off-site protective actions to be implemented for protecting people and their application cannot be limited in terms of times and areas.	Accidents with nuclear fuel melting, resulting in early or large radioactive releases to the environment shall be practically eliminated.
Canada	[III–2]	For operating nuclear power plants (consistent with INSAG-12 [III–3]): Large release frequency (a release of more than 100 TBq of Cs-137)	$< 1{\times}10^{-5}$
		For new nuclear power plants:	
		Large release frequency (a release of more than 100 TBq of Cs-137 or requiring long term relocation of the population)	$< 1{\times}10^{-6}$
		Small release frequency (a release of more than 1000 TBq of I-131 or requiring temporary evacuation of the local population)	$< 1{\times}10^{-5}$
Czech Republic	[III–4]	More than 1% of the initial amount of Cs-137 of the core inventory released to the environment	
Finland	[III–5]	100 TBq of Cs-137	$< 5{\times}10^{-7}$

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
France	[III–6]	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 either contamination of large areas or long-term environmental pollution.	[No quantitative value]
Japan	[III–7]	Taking into account the TEPCO’s Fukushima Daiichi nuclear power plant accident, it is necessary to incorporate the viewpoint of the environmental contamination by radioactive materials into safety goal, and to keep the impact on the environment as low as possible if accidents occur. The frequency of accidents in which the release of Cs-137 exceeds 100 TBq should be reduced to no more than once in one million reactor-years except for those caused by terrorist attacks.	
Russian Federation	[III–8]	The release of radioactive substances into the environment during an accident at a nuclear power plant when, in the 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. It should be noted that the established frequency of release is not a safety goal; it is a safety target.	$< 1\times10^{-7}$

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
Slovakia	[III–9]	> 1% of Cs-137 released from the core inventory	
Switzerland	[III–10]	> 200 TBq of Cs-137 per calendar year	
Ukraine	[III–11]	Large release is defined as requiring public evacuation at the boundary of the protection area.	For existing plants: Criterion: $< 1\times10^{-5}$ Goal: $< 1\times10^{-6}$; For new plants: Criterion: $< 1\times10^{-6}$ Goal: $< 1\times10^{-7}$
United States of America	[III–12], [III–13]	The United States Nuclear Regulatory Commission does not use large release frequency as a safety goal. For new reactor design certification reviews, the Nuclear Regulatory Commission has defined a core damage frequency goal and a conditional containment failure probability goal, complemented by a deterministic containment performance goal. The Nuclear Regulatory Commission uses the large release frequency metric for new reactors of 1×10^{-5} as a screening criterion to inform the staff whether new reactor design applicants are meeting the Commission's expectations for a higher standard of severe accident safety performance and increased margin before exceeding safety limits.	

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
		The Nuclear Regulatory Commission has not approved a formal definition of a large release or a large release frequency. One informal definition for large release frequency is the frequency of an unmitigated release of airborne fission products from the containment to the environment that is of sufficient magnitude to cause severe health effects, regardless of its timing. New reactors transition from large release frequency to large early release frequency metric at or before initial fuel load and discontinue regulatory use of large release frequency thereafter.	

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
Bulgaria	[III–1]	Large releases refers to releases of radioactive material to the environment that necessitate off-site protective actions, which are to be implemented to protect people and the application of which cannot be limited in terms of times and areas. Early releases refers to radioactive releases to the environment that would require off-site emergency measures for protection of the public but that are rendered impossible due to insufficient time to implement them.	Accidents with nuclear fuel melting, resulting in early or large radioactive releases to the environment will be practically eliminated.
Czech Republic	[III–4]	More than 1% of the initial amount of Cs-137 of the core inventory released to the environment within 10 hours after the beginning of the severe accident (temperature of cladding = 1200°C).	
Finland	[III–14]	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. ‘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. The objective is that protective measures are not needed in a situation in which there would practically be no time to implement them.	

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
France	[III-6]	The objective of the design is that event sequences that could lead to large releases with kinetics that might not allow the timely implementation of the measures necessary for the protection of populations should be physically impossible and, if not, very unlikely to happen with a high degree of confidence.	
Hungary	[III–15]	For operating nuclear power plants: a radioactive release in the case of which urgent precautionary measures are required off the site but no sufficient time is available for their introduction	$< 1\times10^{-5}$ for operating nuclear power plants (1×10^{-6} target)
	[III–16]	For new nuclear power plants: (a) Urgent protective measures are required beyond a distance of 800 m from the nuclear reactor OR (b) There is a need for any kind of temporary action (i.e. the temporary evacuation of the population) beyond a distance of 3 km from the nuclear reactor OR (c) There is a need for any kind of subsequent protective measure (i.e. the final resettlement of the population) beyond a distance of 800 m from the nuclear reactor OR (d) There is a need for any long term restriction on food consumption.	$< 1\times10^{-6}$ for new nuclear power plants
Korea, Republic of	[III–17, III–18]	The frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the surrounding population is such that there is a potential for early health effects.	$< 1\times10^{-5}$ for operating nuclear power plants $< 1\times10^{-6}$ for new nuclear power plants

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
Pakistan	[III–19]	The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated.	$< 1\times10^{-6}$
Russian Federation	[III–20]	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 that is prevented by the protective measure and the levels of contamination over a period of 10 days.	The term large early release frequency is not defined in the Russian Federation.
Slovakia	[III–21]	More than 1% of Cs-137 released from the core inventory to the environment within 10 hours after the beginning of the initiating event.	$< 1\times10^{-5}$ for operating nuclear power plants $< 1\times10^{-6}$ for new nuclear power plants
Switzerland	[III–22]	The large early release frequency is the expected number of events per calendar year with a release of more than 2×10^{15} Bq of I-131 per calendar year within the first 10 hours after core damage.	$< 1\times10^{-5}$ for operating nuclear power plants $< 1\times10^{-6}$ for new nuclear power plants

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
United States of America	[III–23]	The large early release frequency 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.	$< 1\times10^{-5}$

Additional information on risk metrics used in other countries can be found in Ref. [III–24].

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24-05216E-T

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ISBN 978-92-0-137424-0

International Atomic Energy Agency
Vienna, 2025
www.iaea.org