IAEA-TECDOC-2042

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IAEA-TECDOC-2042

Optimization of Safety Measures for Protection of Nuclear Installations Against External Hazards

A Framework for the Application of Site Safety Requirements Considering the Safety Features of Nuclear Installations

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OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION OF NUCLEAR INSTALLATIONS AGAINST EXTERNAL HAZARDS

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IAEA-TECDOC-2042

OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION OF NUCLEAR INSTALLATIONS AGAINST EXTERNAL HAZARDS

A FRAMEWORK FOR THE APPLICATION OF SITE SAFETY REQUIREMENTS CONSIDERING THE SAFETY FEATURES OF NUCLEAR INSTALLATIONS

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2024

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Printed by the IAEA in Austria
March 2024

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Optimization of safety measures for protection of nuclear installations against external hazards / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2024. | Series: IAEA TECDOC series, ISSN 1011–4289 ; no. 2042 | Includes bibliographical references.
Identifiers: IAEAL 24-01657 | ISBN 978-92-0-106524-7 (paperback : alk. paper) | ISBN 978-92-0-106624-4 (pdf)
Subjects: LCSH: Nuclear facilities — Safety measures. | Nuclear facilities — Design and construction. | Hazard mitigation..

FOREWORD

Member States have recently made significant progress toward development of advanced reactors including small modular reactors. This new generation of advanced reactors contain safety features quite different from those of traditional large light water reactors. Thus, it is appropriate to review the application of IAEA safety requirements related to site selection, site evaluation, design and safety evaluation for these reactors and to develop additional and/or update guidance to comply with these safety requirements.

An integrated risk management programme provides a framework for implementation of a graded approach consistent with Principle 5 of IAEA Safety Standards No. SF-1, Fundamental Safety Principles. The use of a graded approach based on a 'risk informed' approach is acceptable when the capability of each level of defence in depth is maintained and compliance with the safety objective is demonstrated. The integrated risk management programme provides an approach for the assessment of risk posed by external hazards to nuclear installations that considers additional uncertainties associated with those hazards. For application of a graded approach, the combination of the severity of the site specific hazards used for the design and the design margins needed to meet the target performance goals is identified. The main focus of this approach is to balance the distribution of the margins. This approach is implemented through several iterations to optimize protection of safety measures against site specific external hazards.

This publication provides detailed information on a 'risk informed' and 'performance based' approach for optimization of safety measures for protection against external hazards for advanced reactors with new safety features, with account taken of protection prior to site selection and protection for a site specific application. Considerations regarding the regulatory framework for application of a graded approach are included in the publication. In addition, two examples describing a risk informed approach for design optimization of safety measures against internal and external hazards and application of a graded approach for a site specific probabilistic seismic hazard analysis are presented in the annexes.

In this publication, information is provided to support compliance with the requirements established in IAEA Safety Standards Series Nos SSR-1, Site Evaluation for Nuclear Installations; SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design; and SSR-3, Safety of Research Reactors.

The IAEA would like to thank all the experts who contributed to the development and review of this publication. In particular, the IAEA gratefully acknowledges the contributions to preparation and review of this publication by M. Mahmood (Pakistan), D. Henneke (United States of America) and A. Gürpinar (Türkiye). The IAEA officer responsible for this publication was O. Coman of the Division of Nuclear Installation Safety.

CONTENT

1. INTRODUCTION

1.1. BACKGROUND

The IAEA safety standards provide a technology-neutral framework for the safety of nuclear installations. The IAEA safety standards related to external hazards that might affect the safety of nuclear installations establish requirements and provide recommendations mainly focused on nuclear power plants, with the provision that they can be applied using a graded approach to nuclear installations with lower radiological risk. Paragraph 3.24 of IAEA Safety Standards Series No. SF-1, Safety Fundamentals [1] states:

> "The resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the radiation risks and their amenability to control. Regulatory control may not be needed where this is not warranted by the magnitude of the radiation risks".

The IAEA has published recommendations and guidance on the use of a graded approach for research reactors [2, 3] subjected to external event hazards. The grading is largely based on binning the facility into hazard categories that are a function of the radioactive inventory present in the installation. Assessment of the risk posed by the facility is based on the conservative assumption that, in case of exceedance of the capacity of the installation against a particular hazard, all radioactive inventories may be released. No credit is taken for safety features embedded in the design. Although conservative, but this is unrealistic.

Recommendations on the application of a graded approach for site evaluation for nuclear installations are provided in IAEA Safety Standards [4–6].

The current generation of nuclear power plants and small modular reactors implement advanced safety features which may prove to be rugged against external hazards. Some of the advanced designs may pose a lower radiological risk to people and environment when compared to existing nuclear power plants, therefore, application of the same requirements as for large nuclear power plants would not be consistent with the optimization of safety measures for protection. The challenge is to develop a methodology that allows the application of a graded approach to new designs. Such an approach will need to address other Fundamental Safety Principles, such as Principle 8: Prevention of Accidents, which establishes the concept of 'defence in depth' as the primary means of preventing and mitigating the consequences of accidents [1].

1.2. OBJECTIVE

This publication addresses two main aspects of safety against external events: the site hazard assessment and the robustness of the installation against external hazards. The objectives of the present publication are two-fold:

(1) To contribute to the development of a technology-neutral safety framework for assessing the applicability of site evaluation requirements considering site–installation interactions;
(2) To suggest a methodology for the application of a graded approach and for the optimization of safety measures for protection against external hazards which may take advantage of the robust safety features in the new generation of reactor technology. This approach adopts a risk-informed approach, supporting the effective and balanced implementation of the defence in depth concept.

The overall objective is to present a methodology for the application of a graded approach that credits advanced safety features of the nuclear installation, and allows for an optimized implementation time and cost while ensuring robustness of the installation for applicable hazards. This objective is valid both for new and existing reactors.

This publication may be used by relevant stakeholders in Member States, such as regulatory bodies, operating organizations, vendors, research institutes and technical support organizations working in the area of nuclear safety.

1.3. SCOPE

This publication covers all nuclear installations, new and existing. The benefits of application of the methodology presented herein are expected to be large for advanced reactors prior to construction and for designs incorporating advanced safety features. For existing installations, the benefits will be dependent on the safety features actually implemented in the installation.

1.4. STRUCTURE

This publication is organized in eight sections and three annexes.

Sections 2 to 4 provide background and discussion on framework to support the more specific process for optimization of safety measures for protection against external hazards that are discussed in Sections 5 and 6. Section 2 provides the general framework for optimization of safety measures for protection against external hazards supported by the relevant IAEA safety standards and a risk-informed and performance-based strategy. Section 3 discusses protection for advanced reactor designs with new safety features and Section 4 presents the main elements of a risk informed and performance-based process.

Section 5 discusses optimization of safety measures for protection against external hazards during the design development phase. Section 6 addresses optimization of safety measures for protection against external hazards at the construction license phase including site selection and site characterization considering site–installation interactions. Section 7 presents different regulatory approaches and the use of a graded approach. Section 8 discusses about a possible path forward.

Annexes are also included in this publication. Annex I presents an example of optimization of safety measures for protection against external hazards during the design development of the General Electric (GE) Hitachi PRISM non-light water Small Modular Reactor (SMR). Annex II provides an example of the application of a graded approach for a Probabilistic Seismic Hazard Analysis (PSHA) and comparison between results obtained using a graded approach and Senior Seismic Hazard Analysis Committee (SSHAC) Level 3 results. Finally Annex III provides examples of reactor design characteristics and site characteristics to be considered during the site selection process.

2. GENERAL FRAMEWORK

2.1. GENERAL CHARACTERISTICS OF ADVANCED REACTORS

Advanced reactors being proposed and built today include evolving designs that incorporate new reactor technology that may lead to a lower radiological risk to the public and the environment as compared to older reactors. The design changes and deployment timeline of the nuclear reactors are the basis for the IAEA reactor classification. Early Prototype Reactors (Generation I as per Generation IV International Forum (GIF) terminology) are 'first' reactors developed in the period of 1950-60s. Nearly all Generation I reactors have been shut down and are no longer operating. Generation II (as per GIF terminology) commercial power reactors started operation in 1970s as large power stations with a lifetime of 30–40 years. These reactors may remain in operation until 2030 as some of them have an extended lifetime of 50–60 years. Following the Generation II reactors, the era of advanced reactors started, and those which are currently in operation and deployed after 2010 are known as Generation III advanced reactors (as per GIF terminology). Generation III reactors have robust reactor design and a reduced probability of core damage accidents. The Fukushima Daiichi nuclear power plant accident prompted research and development on the use of passive safety systems resulting in the development of advanced passive designs of nuclear reactors i.e., Evolutionary Reactors (Generation III+ as per GIF terminology). Evolutionary designs have made improvements over current advanced reactors through small modifications. However, the Innovative Reactors (Generation IV as per GIF terminology) have conceptual changes compared to existing designs Prototype testing and/or testing of these innovative reactors through demonstration reactors is needed prior to their deployment for commercial use.

Advanced reactors are typically smaller than existing NPPS. They adopt a standardized design that is easy to install and scale, and that uses advanced technology such as inherent and passive safety features. However, the types of reactors categorized as Generation III+ and Generation IV reactors include a wide variety of fuel types, primary coolant and inherent/passive features. Reference [7] describes over 50 advanced reactor SMR designs both light water reactors (LWR) and non-LWRs, that include reactors designed for power production, steam production or hydrogen production.

Rather than referring to the reactor generation; this publication will refer to both Generation III and Generation IV reactors as advanced reactors. Advanced reactors generally provide a reactor design that has an overall lower risk than previous generations and are designed to be more robust against internal and external hazards. This does not guarantee a low risk for all hazards and may depend on the site specific attributes. For example, a reactor may be designed to be more robust against high winds or aircraft impact by designing much of the plant including the main control room below ground level. If this plant is built in an area with a significant external flood risk, the overall plant risk could be high. However, in general based on current industry practice advanced reactors are designed to withstand a broader spectrum and higher magnitude hazards through implementation of inherent/passive systems that are less susceptible to expected hazards.

External hazards have the potential to damage offsite and onsite power sources, such as the 2011 Great Tohoku earthquake and tsunami, which initiated the long-term station blackout at the Fukushima-Daiichi nuclear power plant. Some advanced LWRs are designed with passive air heat removal that ensures adequate heat removal from the reactor in the event of a loss of offsite power or a resulting station blackout. Passive cooling systems are generally more reliable than active systems following an event and can be built robustly to withstand internal and external hazards. Additionally, passive air heat removal systems as a rule operate without necessity of operator actions. However, passive air heat removal systems are potentially susceptible to events affecting air temperature (e.g. high air temperature) or flow, which need to be analysed for each site. In some designs, use of passive water cooling or injection systems can also operate with high reliability but may require long term operator actions to provide makeup to a storage tank or pool. An example of this is use of an isolation condenser system in the Economic Simplified Boiling Water Reactor, which can operate for up to 7 days without makeup for most scenarios. Other advanced construction techniques (not described here in detail) may be used to ensure the plant is less susceptible to seismic events or geotechnical issues such as soil liquefaction. Many reactors are being designed with the safety systems below grade, and with minimal use of backfill.

Overall, the lower plant risk for advanced reactors, including the improved design against internal and external hazards, can be used in the risk-informed approach and in application of safety requirements to site selection, site evaluation and plant design. This optimization of safety measures for protection against external hazards involves a systematic approach to site investigation, analysis of site specific hazards, and analysis of critical hazards that might have significant contribution to the risk. When significant contributors to risk are identified, regulatory requirements are expected to be established to protect against those hazards. When hazards are determined to be low risk, less extensive regulatory requirements may be possible in consideration with the sufficient margins and defence in depth.

2.2. OVERVIEW OF RELEVANT IAEA SAFETY STANDARDS

Paragraph 3.15 of SF-1 [1] states that "Safety has to be assessed for all facilities and activities, consistent with a graded approach."

Para 3.24 of SF-1 [1] states that "Resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the potential radiation risks."

At the same time, there are societal expectations of a regulatory body around processes to ensure stability and consistency of regulatory control. The reason for this is that society needs confidence that decisions are being made taking into account societal concerns that exist within the regulator's legal mandate. Requirement 26 of IAEA Safety Standards Series No. GSR Part 1 (Rev. 1) [8], Governmental, Legal and Regulatory Framework for Safety states that "Review and assessment of a facility or an activity shall be commensurate with the radiation risks associated with the facility or activity, in accordance with a graded approach".

Requirement 1 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [9] 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.**"

A graded approach is needed to be supported by credible technical evidence such as design and operating experience feedback, or ongoing research and development work. Safety margins need to be well supported, and the risk of cliff-edge effects, as far as possible, be ruled out.

The use of a graded approach is a proportional application of requirements, not a relaxation of requirements.

2.2.1. Site investigation and site safety assessment

Evaluation of site hazard characteristics and physical attributes in accordance with IAEA Safety Standards Series No. SSR-1 [10] is typically linked with the nuclear installation licensing process. Site specific factors associated with installation construction and operations support, risks posed to the nuclear installation and the surrounding area, and the consequences subsequent to a realization of those risks need to be fully identified, analysed, and mitigated as necessary before granting a licence to construct and operate a nuclear installation.

The IAEA has issued Safety Guides providing recommendations on the evaluation of the site specific parameters needed i.e. for design, construction and safety assessment of nuclear installations. These safety standards cover human induced events [11], seismic hazards [4], geotechnical hazards [12], meteorological and hydrological hazards, [13] volcanic hazards [14], and dispersion of radioactive materials in the air, surface water and ground water, considerations for population distribution and feasibility of implementing an emergency plan [15], and other natural hazards [10] like wildfire, drought, hail, river diversion, debris avalanche, and biological hazards. All these safety guides provide recommendations in order to comply with the safety requirements established in SSR-1 [10].

Requirement 3 of SSR-1 [10] states:

> "**The scope of the site evaluation shall encompass factors relating to the site and factors relating to the interaction between the site and the installation, for all operational states and accident conditions, including accidents that could warrant emergency response actions.**"

Paragraph 4.1 of SSR-1 [10] states:

> "The scope of the site evaluation shall cover all external hazards, monitoring activities and site specific parameters relevant for the safety of the nuclear installation. In determining the scope of the site evaluation, a graded approach shall be applied commensurate with the radiation risk posed to people and the environment."

Paragraph 4.2 of SSR-1 [10] states that "The application of the safety requirements for site evaluation for nuclear installations shall be commensurate with the potential hazards associated with the nuclear installation."

Paragraph 4.3 of SSR-1 [10] states that "The level of detail needed in the evaluation of a site for a nuclear installation shall be commensurate with the risk associated with the nuclear installation and the site and will differ depending on the type of nuclear installation."

Paragraph 4.4 of SSR-1 [10] states that "The scope and level of detail of the site evaluation process necessary to support the safety demonstration for the nuclear installation shall be determined in accordance with a graded approach."

The safety standards for site evaluation provide requirements and recommendations for the application of a graded approach for evaluation of external hazards for nuclear installations other than nuclear power plants (NPPs). This application of graded approach is based on potential radiological consequences considering an unmitigated release in the case of an accident.

Recommendations on the application of a graded approach in relation to external hazards for research reactors are provided in Safety Report Series No. 94 [3].

There are two important aspects in the quoted requirements:

(1) The scope of the site evaluation requires to consider factors related to interaction between the site and the installation;
(2) The scope and level of details in the site evaluation process is required to be determined in accordance with a graded approach. Also, the application of the safety requirements for site evaluation is required to be commensurate with the potential radiation risk posed to members of the public and the environment.

This publication is intended to provide a technical basis for application of such a graded approach, considering site–installation interactions. In this way the application of the graded approach for site specific hazards assessment can be better informed regarding the potential likelihood of accidents initiated by a site specific hazard leading to radiological consequences.

In order to be commensurate with the risk associated with the nuclear installation and site specific hazards, an iterative process for optimization of safety measures for protection against external hazards needs to be developed with the following steps:

- Define the parameters of the nuclear installation and the characteristics of the specific site;
- Review the installation design and safety features;
- Perform a conservative (with some simplifications) preliminary site specific hazard evaluation;
- Assess the contribution of the site specific hazard to the overall installation risk;
- Enhance the site specific hazard assessment if the risk contribution is significant for that hazard.

A simplified preliminary site specific hazard assessment may use limited regional and site specific investigations leading to higher uncertainties compared to a full scope hazard assessment. In addition, a simplified approach using limited data for hazard assessment generally provides more conservative results as compared to the full-scope hazard assessment.

When using simplified hazard assessment methods (including use of more conservative hazard parameters) to demonstrate that the site specific hazards represent a low risk for the installation, it is still necessary to demonstrate the design adequacy. This may necessitate a greater safety margin in the design against the site specific hazard to compensate for conservatism and higher uncertainties, including a more robust set of design requirements. However, if the hazard contribution is small, the safety margins may be sufficiently large that may to not need a higher robustness in the design. The benefits are mainly related to reducing the time and cost for the site specific hazard evaluation.

In view of the above, some of the safety features of the advanced reactor design can be credited for evaluation of the consequences and application of the graded approach.

2.2.2. Design of a nuclear installation against external events

For NPPs, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [16] establishes requirements related to design against external and internal hazards.

Requirement 10 of SSR-2/1 (Rev. 1) [16] states:

> "**Comprehensive deterministic safety assessments and probabilistic safety assessments shall be carried out throughout the design process for a nuclear power plant to ensure that all safety requirements on the design of the plant**

are met throughout all stages of the lifetime of the plant, and to confirm that the design, as delivered, meets requirements for manufacture and for construction, and as built, as operated and as modified."

Paragraph 4.17 of SSR-2/1 (Rev. 1) [16] states:

> "The safety assessments shall be commenced at an early point in the design process, with iterations between design activities and confirmatory analytical activities, and shall increase in scope and level of detail as the design programme progresses."

Requirement 17 of SSR-2/1 (Rev. 1) [16] states:

> "**All foreseeable internal hazards and external hazards … shall be identified and their effects shall be evaluated. Hazards shall be considered in designing the layout of the plant and in determining the postulated initiating events and generated loadings for use in the design of relevant items important to safety for the plant.**"

Paragraph 5.21of SSR-2/1 (Rev. 1) [16] states:

> "The design of the plant shall provide for an adequate margin to protect items important to safety against levels of external hazards to be considered for design, derived from the hazard evaluation for the site, and to avoid cliff edge effects."

Paragraph 5.21A of SSR-2/1 (Rev. 1) [16] states:

> "The design of the plant shall also provide for an adequate margin to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design, derived from the hazard evaluation for the site."

Requirement 20 of SSR-2/1 (Rev. 1) [16] states:

> "**A set of design extension conditions shall be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments for the purpose of further improving the safety of the nuclear power plant by enhancing the plant's capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures. These design extension conditions shall be used to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences.**"

Paragraph 5.27 of SSR-2/1 (Rev. 1) [16] states:

> "The plant shall be designed so that it can be brought into a controlled state and the containment function can be maintained, with the result that the possibility of plant states arising that could lead to an early radioactive release, or a large radioactive release is "practically eliminated".

The possibility of certain conditions that may lead to uncontrolled large releases may be considered to have been 'practically eliminated' if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise.

2.2.3. Application to advanced reactors

Advanced reactor designs need to provide a combination of active, passive and inherent safety features that contribute to the effectiveness of the physical barriers in confining radioactive material at specified locations and may lead to practical elimination of severe accidents with significant radiological consequences.

Passive safety systems could be challenged by the external hazards and could potentially contribute to large or early accident sequences. The design is required to be robust to ensure protection against external and internal hazards impacting such passive safety systems.

The risk metrics used to measure the performance of the design could be core damage frequency (CDF), Large Early Release Frequency (LERF), Large Release Frequency (LRF) or the resulting source term.

Very low values for CDF, LERF/LRF and/or a sufficiently small source term related to potential accidents initiated by external hazards imply a low contribution of a given site specific hazard to the installation. Results of safety assessment of the design showing a small source term or very small CDF, LERF or LRF will provide a basis for application of the graded approach.

2.3. OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION AGAINST EXTERNAL HAZARDS BASED ON RISK-INFORMED AND PERFORMANCE-BASED APPROACH

Optimization of safety measures for protection in design generally covers many aspects. In this publication, the focus is on optimization of safety measures for protection against external hazards presented for two distinct phases:

(1) The design development phase based on generic (bounding) hazard parameters;
(2) The construction licensing phase for a site specific installation.

Generic site parameters utilize generic and industry data, consistent with the assumption in the risk assessment, to provide conservative or bounding site parameters for each applicable hazard. The generic site parameters selected need to be bounded for all of the potential sites (for each hazard) where the design may be constructed. Although the site parameters are based on generic data, the selected parameters would be a bounding set for the site. The generic (bounding) parameters are not meant to represent a specific site, as the combinations of hazards would likely not be present in a single site. However, as noted in Section 5.3, if the generic (bounding) site parameters are too conservative, limitations may be placed on the site in response to a more limiting set of site parameters (e.g. limiting the sites to low or moderate seismic hazard sites).

The optimization of safety measures for protection against external hazards process in the scope of this publication is based on a risk-informed and performance-based (RIPB) approach and includes considerations about the enhancement of defence in depth as a result of this process. The process described in the sections below generally applies a RIPB approach in support of applying a graded approach, and the application of a RIPB approach in either modifying the design, site evaluation or licensing process is described as the “process for optimization of safety measures for protection”.

One important objective is to achieve robust defence in depth and in particular applicable to advanced reactor designs in relation to challenges produced by external hazards.

The key elements of the process for optimization of safety measures for protection are:

- The application of the relevant safety provisions (for site evaluation, design and safety assessment) shall be commensurate with the potential hazards associated with the nuclear installation. This can be done using an adequate graded approach process;
- The use of a risk-informed approach to fully utilize the insights from systematic risk assessment in combination with structured prescriptive/deterministic rules. This approach can provide reasonable assurance that protection is provided against radiological risk to public;
- The use of a performance-based approach to evaluate effectiveness relative to realizing achievable desired outcomes by using quantifiable performance targets (e.g. frequencies of accidents or of undesirable consequences, performance requirements for structures, systems, and components (SSCs) capabilities to prevent and mitigate accidents, etc.).
- Continued use of the defence in depth principle to ensure a robust and resilient installation.

In consideration of the above, it is important that the total plant risk is considered, both in ensuring that the plant achieves the performance targets for all hazards including combination of hazards and initiating events, as well as ensuring that unintended consequences do not result from any of the activities of the process for optimization of safety measures for protection. For example, reductions in thickness of a structure as a result of low plant risk for missiles or high winds might result in changes in the seismic fragilities for a building and an increased seismic risk. Overall, optimization of safety measures for protection involves an integrated analysis for all possible hazards, hazard combinations and initiating events.

Optimization of safety measures for existing nuclear power plants already in operation would involve similar processes, but different steps as discussed above. For example, the refinement of the hazard analysis as well as improvements to the design could be performed in a similar fashion. However, given the site is already evaluated, the optimization of safety measures for a site specific application is not worthwhile.

One aspect of optimization of safety measures for protection against external hazards not fully discussed in this publication is the effect of climate change on the process of optimization of safety measures. Climate change effects the possible frequency and magnitude of potential external events affecting a reactor at a specific site, including the increased likelihood for combinations of hazards such as a combined high wind and extreme rain event. The current nuclear sites are already measuring the changed environmental conditions, for example resulting in increased rains, higher temperatures and increased flood levels. As a result, during the licensing process, the design would generally be analysed for the potential range of external conditions and hazards that may occur during the lifetime of the plant. In some cases, designing a plant against potentially higher winds, higher and lower temperatures and other environmentally affected conditions can be costly and significantly affect the design optimization of safety measures process. The evaluation of potential climate change does require increased data collection during the site evaluation process.

Each reactor design potentially has different vulnerabilities which may be triggered by different external hazards or conditions, including advanced reactors using passive systems such as passive air cooling. Both passive and active systems may be affected by consequences of climate change, which would need to be considered during the optimization of safety measures process. This process needs to be performed using realistic analysis, with the consideration of uncertainty that estimates the upper and lower bound estimates for the increased hazards, hazard combinations and environmental conditions.

2.3.1. Design development phase

This publication does not address the process for selecting the generic site related parameters (site independent hazards) that are selected to be considered as input for the safety assessment of the design.

Paragraph 4.17 of IAEA SSR 2/1 (Rev. 1) [16] states:

> "The safety assessment shall be commenced at an early point in the design process, with iterations between design activities and confirmatory analytical activities, and shall increase in scope and level of detail as the design programme progresses."

In the design development phase, the objective of the optimization of safety measures is to increase safety performance of the design in general and in relation to internal and external hazards in particular. This publication addresses mostly safety performance of the design against external hazards and means for optimizing safety measures for protection against external hazards. One aspect of the optimization of safety measures process in the design stage is that it is not desirable to perform significant design changes for site specific applications, especially if the design is being applied to multiple sites. This can be supported, in part, by the use of bounding hazard estimates for the range of sites being considered, as discussed in Section 5.1. Additionally, the performance of a more detailed design phased assessment can result in better optimized safety measures for protection at this phase (such as relaxation of design requirements), rather than potential optimization of safety measures for the site specific application, which may result in site-to-site variations for the same generic design.

Optimized safety performance for nuclear installations is required to achieve an adequately robust design against external hazards, e.g. adequate safety margins, prevention of common cause failures, sufficient independence of the defence in depth barriers, low risk contribution to accident sequences initiated by external hazards considered in the design, etc.

One challenge to optimization of safety measures for protection during the design phase, especially early in design, is the availability of design information and design details that are key to the protection against external hazards. The initial design is generally provided with consideration for potential internal events, such as loss of cooling, loss of coolant accidents (LOCA), etc., but may be provided without consideration for external hazards. This challenge can be mitigated in part through the use of assumptions and a decision tracking, which may result in detailed design requirements if the assumptions affect the risk and goals of optimization of safety measures for the installation. As the design is finalized, these assumptions and requirements can be confirmed or updated, which may require review of the optimization of safety measures process and results if the assumptions are changed.

Optimization of safety measures for protection against external hazards during the design process is required to achieve safety margins for potential hazards and ensure the practical elimination of sequences that contribute to an early or large radioactive release (see for example, Requirements 5 and para. 5.37 of SSR-2/1 (Rev. 1) [16].

2.3.2. Construction license stage for an installation on a specific site

This stage corresponds to the situation when nuclear installation design based on bounding hazard parameters for a nuclear installation is complete, a site has been selected and the process for obtaining a construction license for the installation at that site has started. The objectives of optimization of safety measures at this stage are similar to the design stage (based on bounding hazard parameters), with a focus on ensuring acceptably low radiological risk for each site specific hazard while minimizing overall burden. However, at this stage it might not be feasible to look for potential optimization of safety measures that might result in significant design changes that may necessitate going back to the design certification stage.

The objectives for optimization of safety measures at the construction license phase include the following:

(a) When analysing site specific hazards during the construction license phase (for installation on specific site), the hazard analysis performed during the design phase would need to be confirmed. This design-phase analysis may have used either a deterministic or graded approach but may not bound the site specific hazards. This can include the case where the design did not consider a hazard potentially important for a specific site, which would not be analysed for the first time during the construction phase. Also, it is likely that for the site specific conditions; the difference in severity of the hazards used for generic design and site specific hazards may provide additional safety margins in respect to those hazards which can be leveraged to reduce analysis or design requirements. At this stage the objective of optimization of safety measures is to reduce the effort and time allocated for site characterization;

(b) To optimize safety measures for the implementation of requirements for hazard qualification specified for suppliers of equipment and components.

The first objective can be achieved by using a graded approach process for hazards characterization and confirming that the conservative hazards estimates will not lead to significant risk contribution.

The second objective can be achieved by performing a preliminary conservative simplified hazard assessment and demonstrating that the hazard is not a risk significant contributor (e.g. that the equipment qualification requirements are sufficient).

2.3.3. Optimization of safety measures for protection against external hazards for existing nuclear installations

As discussed, the optimization of safety measures process can be applied to existing facilities (e.g. when a new hazard evaluation is performed). Of course, an existing facility with an established deterministic or probabilistic hazard analysis would generally not need further analysis. However, if further analysis is needed when considering the establishment of new requirements, such as following the accident at the Fukushima Daiichi nuclear power plant [17], a graded approach can be applied. The analysis may rely on the existing plant analysis unless the hazards or combinations of hazards are being analysed for the first time for the existing plant – such as a combined seismic event followed by a tsunami. In general, the methods given in Section 6 would be used for application of new analysis for existing reactors.

2.4. DEFENCE IN DEPTH CONSIDERATIONS

As discussed in Ref. [16] defence in depth is implemented primarily through the combination of a number of consecutive and independent levels of protection that are designed to protect the public and the environment from potential radiological exposure. If one level of protection or barrier were to fail, the subsequent level or barrier would be available. The independent effectiveness of the different levels of defence is a necessary element of defence in depth. Refs [16, 18] describe five defence levels (DLs), which are not discussed in detail in this publication.

The defence in depth adequacy and design provisions after the establishment of an adequate design margin optimization of safety measures for protection against external hazards need to be checked. The process considers all potential challenges and challenging mechanisms related to external hazards for assessing the defence in depth attributes (such as independence of the barriers, sufficient margins and resilience against potential failures of each barrier, etc.) and for assessing if design provisions adopted to cope with all challenges to the defence in depth barriers are adequate for all defence in depth levels.

Defence in depth and other deterministic considerations are necessary to be considered and complemented by risk and performance-based considerations supporting the risk informed decisions in relation to the process of optimization of safety measures for protection.

To achieve a robust and balanced defence in depth provides part of the basis for demonstrating that the safety objectives are met.

2.5. ROLE OF STRUCTURES, SYSTEMS AND COMPONENT SAFETY MARGINS

The SSC safety margins play an important role in the development of the SSC design requirements for reliability and performance capability. Acceptance limits on the SSC performance are set with safety margins between the level of performance that is deemed acceptable in the safety analysis and the level of performance that would lead to damage or adverse consequences for all the postulated initiating events in which the SSC performs a prevention or mitigation function.

The magnitudes of the safety margins in performance are set considering the uncertainties in performance, the nature of the associated hazards considered in the design, and criteria for adequate defence in depth. The ability to achieve the acceptance criteria in turn reflects the design margins that are part of the SSC capability to mitigate the challenges associated to external hazards.

The reliability targets are set to ensure that the underlying postulated initiating events frequencies and consequences meet the applicable safety evaluation criteria with sufficient margins. These safety margins are also assessed in the evaluation of defence in depth.

The evaluation of margins between the frequencies and consequences of external events and the frequency-consequence target curve (if used) and the margins between the calculated cumulative risk metrics and the site risk targets are evaluated in the safety assessment for the design. The consideration for sufficient margins during the evaluation of hazard protection process is discussed further in Sections 4–6.

3. OPTIMIZATION OF SAFETY MEASURES FOR ADVANCED REACTORS WITH NEW SAFETY FEATURES

3.1. GENERAL

As discussed in Section 2.1, advanced reactors employ a range of features, fuel types, cooling types and applications. General characteristics that affect optimization of safety measures for protection against internal and external hazards include the following:

- Inherent Design Features: The term inherent design feature is used broadly to describe a range of features such as use of ceramic fuel to designs where the coolant covers the fuel during all accident conditions within the design basis and design extension conditions. The use of ceramic fuel is more similar to a passive design feature. For this situation, the features are categorized in two general ways:
    - Inherent features not affected by hazards or accident sequences. An example would be use of TRi-structural ISOtropic fuel, where the coating does not allow the release of significant radionuclides until the temperature exceeds a certain threshold [7]. In this example, the TRi-structural ISOtropic fuel acts like a primary barrier against releases but it is not failed during events such as a high magnitude seismic event. Other features may include inherent reactivity feedback (e.g. reduced reactor power as the temperature goes up), absorption by the primary coolant of any radioactive release from damaged fuel[1] (e.g. for sodium cooled reactors), and designs with significant thermal heat capacity (e.g. reactors can withstand full loss of cooling for long periods of time without fuel damage).
    - Inherent features potentially affected by hazards or accident sequences. Similar to passive systems, these would include features where an external hazard can impact the reliability or capacity of the feature. One example is the use of large volumes of water in the DHR400 [7] which are potentially affected by external hazards.
- Passive Design Features: Ref. [19] defines four categories (A to D) of passive systems depending on whether there are external signals to initiating, external power to initiate, moving parts to the system and the need for moving fluid systems. These categories are generally applicable to all reactor types with passive systems, with the understanding that inherent design features discussed above may be categorized as a passive system. Generally, passive systems are highly reliable and robust, but are potentially impacted by internal or external hazards. The impact can be mitigated, as most passive systems are designed to be fail safe.
- Active Design Features: Active systems include a majority of safety systems relied upon for the current generation of nuclear power plants. The robustness of each system against internal and external hazards is highly dependent on the system response, including required support systems such as electrical power, during an accident.
- Required Safety Functions (RSFs): These functions include for all reactors the control of reactivity, heat removal, and confinement (or control of accident releases). NPP design can envisage additional safety functions that are intended to support mentioned

[1] Although the absorption of radioactive material is un-effected by internal and external hazards, the function can be lost given the primary coolant is lost – such as following a failure of the reactor vessel.

fundamental safety functions (such as primary pressure control, primary level control and power supply etc.)[2].
- Reactor Power and Potential Source Term: As discussed in Ref. [7], the range of proposed reactor designs is large, ranging from portable reactors with a small power output (e.g. below 1 MW(e)) to similar power levels of reactors currently in operation. When considering the fuel type (discussed above under inherent features), as well as the normal operating configuration (e.g. pressure, chemical form of radionuclide transport), the potential source term for many SMRs can be very small.
- Multi-unit or Multi-module operation: Many SMRs with smaller power generated from an individual plant may be designed to operate on a site with several or many reactors of the same design. The resulting source term can be increased if the initiating event results in a release from multiple (or all) reactors on a site.

When categorizing the response of the nuclear installation to specific hazards, the initial assessment is what RSFs are necessary, what type of design feature is protecting the installation against the hazard, and whether the design feature is affected by the hazard. An example would be the pool-type reactor (no pressure or level control required) using sodium cooling (absorbs most fuel-damage releases), Inherent Reactivity Feedback (IRF) credited to control reactivity, and passive air cooling around the vessel. For this example, when looking at a seismic event, the susceptibility of the reactor design is generally limited by the seismic capacity of the passive air cooling system. Optimization of safety measures for protection against external hazards is therefore relatively simple, and centred, in general, around the protection of the passive air cooling. In another example, a Gen-III LWR, where RSFs are provided by a series of active systems; the optimization of safety measures for protection against external hazards becomes more complicated – requiring detailed analysis of accident scenarios related to each RSF.

The optimization of safety measures to protect against external hazards can be influenced by whether the site has been selected and whether the analysis of the response of the installation to specific hazards is performed against site specific hazards or using bounding site parameters. The optimization of safety measures is also influenced by the resulting risk estimate if performed using bounding site parameters versus site specific parameters. If for example, a bounding hazard is used for the high winds probabilistic safety assessment (PSA), and the resulting risk estimated from the PSA is low (e.g. high winds is not risk-significant), in this case when the site is selected where the site specific wind hazard is lower than the bounding hazard; the level of effort for both the site characterization and the high winds PSA can be reduced. This is discussed further in this publication.

The following sections provide the objectives and proposed process for optimization of safety measures for protection against external hazards, which accounts for both the range of reactor design features discussed in Section 2.1 and the safety functions that need to continue to be performed during each external hazard event.

3.2. OBJECTIVES OF OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION AGAINST EXTERNAL HAZARDS

The optimization of safety measures for protection against external hazards may target the following objectives:

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[2] Many NLWR advanced reactors are pool-type reactors (atmospheric pressure), and do not need primary pressure and level control functions following an event (for reactors where the primary coolant does not leave the vessel).

(1) During the design development based on bounding hazard parameters:
   - Assumptions: Bounding site parameters are used as input to design. Design is under development and a series of safety assessments of the design are performed (few iterations) as the design is progressing and more details are available.
   - Achieving acceptable risk contribution related to external hazards based on bounding site parameters. Safety assessment of the design is conducted to conform that the design provisions are adequate and the risk results are in acceptable range.
   - Achieving adequate design robustness against external hazards (e.g. sufficient margins to avoid cliff edge effects and sufficient margins for preventing large early releases).
   - Achieving a balanced risk distribution which considers safety margins for all possible external hazards. Where the risk results are unacceptable or where the margins can be improved, design robustness may be improved. Where the risk results and safety margins are sufficient, design requirements may be reduced.
   - Implementing the above objectives in a graded approach which allows for reduced time/cost of implementation, while ensuring design robustness for each applicable hazard.

(2) During the licensing process for nuclear installation on a specific site:
   - Assumptions: Detailed design based on bounding hazard parameters is completed, the site was selected, and site characterization is in progress.
   - Achieving adequate site specific robustness against site specific external hazards (e.g. sufficient margins to avoid cliff edge effects and sufficient margins for preventing large early releases).
   - Ensuring that optimization of safety measures of the overall effort for qualification of the SSCs is commensurate with radiological risk and site specific hazards.

3.3. OVERVIEW OF THE PROCESS OF OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION AGAINST EXTERNAL HAZARDS

As discussed in 2.3.2, the general approach for designing a nuclear installation against external hazards includes:

(1) The analysis to select, define and screen the plant design basis external hazard (DBEHs) events;

(2) The analysis and documentation that demonstrates a high confidence for the SSCs necessary to achieve a controlled state and then safe state are available and have a very low likelihood of failure for DBEHs. This step includes potentially the optimization of safety measures and consideration of margins and includes analysis for both the site characterization and the plant response for a range of hazards including DBEHs and beyond design basis external hazards. The process of optimization of safety measures has to allow for a reduced analysis effort when the hazard risk has sufficient margins and acceptably low risk (when analysed with a simplified/conservative approach);

(3) Establishment of requirements for the SSCs necessary to achieve a controlled state and then safe state for DBEHs and beyond design basis external hazards, including the assessed defence lines and design requirements. Where the hazard analysis demonstrates sufficient margins and low risk for a specific hazard, the SSCs requirements may be reduced or eliminated.

The process of optimization of safety measures discussed in Sections 5 and 6 is performed using RIPB approaches on the three basic steps discussed above, that include supporting sub-steps, such as the safe shutdown equipment list selection performed for each DBEH. The key to the optimization of safety measures approach is in (2) above, which includes the consideration for both the site characterization as well as the safety analysis of the installation.

3.4. CONSTRAINTS FOR THE OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION

The optimization of safety measures for the protection of a nuclear installation against external hazards involves both the reduction of burden, where possible, and where needed for higher risk hazards, and the increase in completeness in the analysis over the existing deterministic approach. As far as the second step, this involves performance of additional risk-informed analysis (over a traditional deterministic analysis). The hazard PSA, as discussed in 4.2.1, to inform the existing site deterministic hazard analysis can be performed for any site or design as a risk reduction (or confirmation of low risk) step.

As far as reduction of burden, the overall goal of the process of optimization of safety measures for protection is to reduce burden while maintaining safety goals. As a result, any reduction in burden (or requirements) would be difficult if the risk from the nuclear installation for the hazard being considered is high, or if the overall risk from the nuclear installation is high for all hazards/causes. Note that optimization of safety measures including the use of a risk-informed approach does not directly compare "burden" with risk changes, since they are different measures. The burden is one factor when analysing a potential plant change, including design or analysis changes potentially involving regulatory burden. For example, a change which resulted in a very small (and acceptable) increase in risk but had only a small reduction in burden would not be as attractive as a change resulting in very small increase in risk resulting in a significant reduction in burden. Additionally, even though a change can significantly reduce burden if the risk increase is unacceptable; the change would not be acceptable.

Ref. [20] provides a framework for making changes to a plant licensing basis. The acceptance guidelines in Ref. [20] include a consideration for the plant overall risk and indicate that small risk increases need not be considered if the plant risk is above established national regulatory target values for CDF or LERF: for example, Ref. [20] sets a value of 1E-04/year for CDF and 1E-05/year for LERF for existing reactors. The general concept is applicable to other countries, where the site overall risk is measured against quantitative measures. Note that Ref. [20] allows for licensing changes that are risk reductions at any time, regardless of plant overall risk. Finally, any use of PSA for reducing burden need to consider the completeness and uncertainty of the PSA, which includes consideration of the completeness of the PSA, the level of conservatism and consideration of "masking" of the risk insights due to over conservatism in the analysis.

As an example of completeness, if a site has performed a simplified seismic PSA but the scope did not consider seismically induced LOCA sequences, then a reduction of burden on systems that may respond to LOCA events would not be supported. An example of masking would include a site PSA, where the dominant hazard is estimated using an extremely conservative assessment, and a less important hazard is initially estimated as not risk-significant. When considering the uncertainty, if the conservative hazard is estimated realistically, it may be the case that a less important hazard can become a significant contributor. In this case, burden reduction may still be possible if the overall site risk is very low; but the support for the change would need to consider the uncertainty introduced by the overall state of the PSA.

In some cases, such as for risk-significant scenarios, it may be better to fully characterize and quantify the hazard in order to take full advantage of a risk-informed process of optimization of safety measures. There are several aspects affecting this consideration. First, as discussed above, a limited scope evaluation is generally conservative. If the conservatism results in a risk estimate that is unacceptably high, options for optimization of safety measures for protection may be limited. Second, a limited scope evaluation may not provide the insights needed to fully understand the optimization of safety measures options. These options may be either masked by a conservative but dominant risk scenario or may not be modelled in sufficient detail to provide designers with the information need to support optimization of safety measures for protection. Additionally, with a conservative assessment, key steps performed in a risk-informed process, discussed in Section 4.2, might not be able to be performed, such as establishing performance goals, determining functional requirements, etc.

Note that if qualitative analysis (including screening) or quantitative analysis for a hazard is not possible, the optimization of safety measures for the design for protection against that hazard would be difficult.

4. THE RISK-INFORMED AND PERFORMANCE-BASED APPROACH

4.1. GENERAL CONSIDERATIONS

A framework for an integrated risk-informed decision making (IRIDM) process is described in Ref. [21], which includes a description of RIPB processes.

In accordance with Ref. [21], IRIDM is a systematic process which integrates all of the major considerations influencing NPP safety. The main goal of IRIDM is to ensure that any decision affecting protection of nuclear safety measure is optimized without unduly limiting the conduct of operation of the NPP. It underpins nuclear safety decisions and ensures consistency with the safety goals of the Member State. The outcome of IRIDM needs also to satisfy the following principles:

- Defence in depth is maintained;
- Safety margins are maintained;
- Engineering and organizational good practices are taken into account;
- Insights from relevant operating experience, research and development, and state of the art methodologies are taken into account;
- Adequate integration of safety and security is ensured;
- Relevant regulations are met.

Protection against external hazards without due considerations of deterministic and probabilistic aspects can result in inconsistency and uncertainty in protection. On the other hand, application of deterministic seismic requirements, which are applied using conservative margins, can result in significant costs that may not be proportional to the benefit, or risk reduction. Considering the lessons learned from the accident at the Fukushima Daiichi NPP, para. 5.21 of IAEA SSR2/1 (Rev. 1) [16] states:

> "The design of the plant shall provide adequate margins to protect items important to safety against levels of external hazards to be considered for design, derived from the hazard evaluation for the site".

This requirement ensures margins to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design. The general considerations for defence in depth and margins are discussed in Sections 2.4 and 2.5, and the specific considerations in evaluation steps are presented in this section.

The purpose of an RIPB approach is to both provide reasonable assurance that adequate protection is provided for the public, and to ensure the requirements applied are proportional to the risk, accounting for uncertainty.

The risk-informed approach involves the performance of a risk assessment (e.g. the PSA) in combination with a structured deterministic approach which addresses the uncertainties from the risk assessment. When reducing the burden from an existing requirement, the risk assessment insights (including importance measures) are used to revise the application of established requirements while maintaining the existing safety margins and defence in depth [20, 22–23]. When providing an alternate approach to an existing deterministic approach, a risk-informed approach involves the application of deterministic rules that provide reasonable assurance of adequate protection. This structured deterministic approach depends on the application but involves the consideration of both the uncertainty and assurance of adequate defence in depth. Uncertainties are sometimes difficult to quantify. As far as possible, the strength of knowledge of the models (including assumptions, simplifications, data and phenomena understanding) are to be assessed.

An approach that is performance-based involves an approach that establishes performance and results as the primary bases for decision making [24]. Performance-based regulations have the following attributes:

- Measurable, calculable, or objectively observable parameters exist or can be developed to monitor performance.
- Objective criteria exist or can be developed to assess performance.
- Licensees have flexibility to determine how to meet the established performance criteria in ways that encourage and reward improved outcomes.
- A framework exists (or can be developed), in which the failure to meet a performance criterion, while undesirable, will not result in an immediate safety concern.

When an approach is RIPB, this describes a wide variety of applications that mix both risk and measurable goals in the application. The range may involve the following possible approaches:

(1) An approach involving use of both risk insights for selected requirements and performance measures for other requirements [25];
(2) An approach involving acceptable risk measures, such as use of quantitative health objectives (QHOs) [26].

The application of a RIPB approach in this document is referring to the second application above.

When considering RIPB approaches for advanced reactors, several factors affect the approach:

(1) Advanced reactors generally are designed to have a significantly lower risk than existing plants for both internal and external hazards.
(2) Generally, the regulatory intention is to build advanced plants with a significantly lower risk than existing plants.

(3) Advanced reactor designs, including SMRs, have significant variance [7] in source term, plant features, protective measures and protection against external hazards. Protective measures include a range of inherent features (e.g. ceramic fuel), passive safety systems, and other design features that make overall plant risk significantly lower than the existing fleet.
(4) In general, advanced reactors are simpler in design and have less components.

As a result, an approach for protection against external hazards has to be flexible enough to be applied to a wide variety of plant designs.

The graded approach for advanced reactors would also be greatly impacted by the design features discussed in Section 2.3.1. This includes the following considerations:

(1) Plants using inherent or passive design features, especially where the features are robust against potentially significant external hazards, can be addressed more easily using risk analysis to screen hazards or analyse the hazards using a simplified approach. Several examples are provided in Section 2.3.1. However, as noted, if the feature is potentially impacted by a hazard, then a detailed hazard assessment may be needed. Use of inherent and passive design features generally results in both a simplified design (e.g. less safety related systems and SSCs) and a resulting simplified analysis including the PSA. When the risk-significant functions and supporting SSCs have a robust design against the expected plant hazards, these design features can support either screening the hazard from the deterministic analysis or use of a simplified approach.
(2) Installations with multiple units or multiple modules on the same site need to consider the interaction between units when addressing external hazards. This includes the interaction between units and use of shared systems, control rooms and operating personnel, as well as the expected release that may occur with multiple releases give a multi-unit or multi-module core damage event. Ref. [27] demonstrated that in many cases, multi-unit risk is dominated by accident sequences involving external hazards, such as seismic events.

The general RIPB approach for advanced reactors in the United States of America (USA) is provided in Ref. [26]. An example application of risk-informed approach is discussed in Annex I. Performance-based codes, such as Refs [28, 29] are utilized to develop an example process. Ref. [30] describes this process. This process can be used for both bounding and site specific designs.

4.2. IMPLEMENTATION DETAILS

Two types of methodologies are generally available for safety assessment against external hazards:

- Deterministic safety assessment (DSA) aimed at evaluating the failure capacity of the success path. The success path is defined by the availability of the SSCs required to perform the fundamental safety functions and to bring and maintain the NPP into a controlled state and then into safe state.
- RIPB (graded) approach aimed to evaluate the contributions to all possible accident sequences and scenarios induced by external initiating events.

Establishing the design basis and beyond design basis hazard levels applied to the DSA may be supported using the PSA information to determine annual exceedance frequencies, without the performance of a comprehensive PSA.

The following sections provide details of assessing a hazard using a PSA, although many of the qualitative steps of the PSA are the same or similar to the deterministic analysis steps. However, where there are conflicts, this is noted. For example, in the screening steps, it is possible to screen a hazard in the deterministic hazard analysis, but need detailed analysis in the PSA.

4.2.1. Probabilistic safety analysis

The methodology and attributes used to develop a PSA for an advanced reactor or SMR is generally the same as a PSA performed for the other types of reactors (e.g. existing LWRs). The recommendations on PSA are described in relevant IAEA safety standards [23, 31], supporting IAEA technical publications (e.g. Ref. [22]) and other international standards (e.g. Ref. [32]). The PSA methodology described in these standards and publications, including Refs [23, 31] are generally applicable to advanced reactors and SMRs, including both Advanced LWRs and Non-LWRs (NLWRs), the high-level PSA technical elements are described below. The following are the general steps in the development of a comprehensive PSA:

- Scope of the PSA;
- Selection of PSA risk metrics;
- Identification of operating states to be modelled in the PSA;
- Identification of initiating events;
- Analysis of accident sequences and success criteria;
- System reliability analysis;
- Human reliability analysis;
- Data analysis and common cause failures analysis;
- PSA quantification and interpretation of results;
- Internal hazards analysis;
- External hazards analysis;
- Level 2 PSA and Level 3 PSA, when performed;
- Multi-units PSA (MUPSA);
- Mechanistic source term and radiological consequence analysis;
- PSA applications.

The scope of the PSA needs to be consistent with the nuclear installation and the national safety goals or criteria (if set), similar to a PSA for LWRs. As discussed in IAEA Safety Standards Series No. SSG-3 (Rev. 1), Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [23], the high level, quantitative results of the PSA are often used to confirm compliance with safety goals or criteria, which are usually formulated in terms of quantitative ranking from CDF, frequency of radioactive releases, and societal risk from release of nuclear material (see these risk metrics discussed in more detail in Section 4.2.2).

Reference [22] provides attributes for a full scope Level 1 PSA [23]. The applicability of the attributes to various reactor types is discussed in Section 1.6.1 of Ref. [22]. Reference [22] does clarify that the overall scope of hazards and PSA attributes, such as performance of initiating event analysis, accident sequence analysis, etc. is applicable to all reactor types. As such, the scope of analysis and hazards for SMRs, which is presented in Sections 6.2–6.14 is similar to a

PSA for existing reactors. Each section clarifies technical areas where the attributes or analysis may be unique for SMRs, starting with the development of PSA risk metrics discussed above.

Section 1.6.2 of Ref. [22] also discusses the applicability of PSA for different stages of design, and states that "it is recognized that significant differences exist regarding PSA approaches, modelling aspects, and data for different stages of the plant's lifetime."

Similar differences are expected for SMR PSAs, potentially exacerbated by being first of a kind or new designs where extensive testing, commissioning or operating experience is not available.

4.2.1.1. Screening of a Hazard from the PSA

In the steps discussed above, a complete and thorough analysis of all applicable hazards to a site or design is needed in order to meet the IAEA safety requirements and national PSA guidance. Once a potentially applicable hazard is identified, the initial steps are to reduce the detailed analysis by either qualitatively or quantitatively screening of low risk hazards or hazard groups from further consideration. The required steps are discussed in Ref. [22], under the Hazard Screening and Final Hazards List Identification attributes or in the U.S.A. PSA standard under the screening criteria SCR-1, 2 and 3. In both cases, screened hazards are demonstrated to be not risk-significant by the process and review performed during screening.

Unscreened hazards are quantitatively analysed in detail. Hazards analysed in detail may or may not be risk-significant, similar to other plant initiating events. In other words, hazards included in the detailed PSA may still be found to be low risk and addressed in the process in a similar manner to the screened hazards.

The specific steps for screening a hazard either prior to or after a site is selected are as follows:

(1) Identify a list of potential hazards affecting the plant. The potential list of hazards is as comprehensive as possible, considering bounding and potential site specific hazard sources. Hazards may be grouped during the identification process. A hazard group involves hazards with similar effects or challenges to the facility that can be assessed using a common approach. Development of the list of hazards, which potentially affect the generic design or are site specific, is key to the process of optimization of safety measures. If the list of hazards does not include all potential hazards, consideration for the missing hazards during the licensing process can result in increased costs involving redesign, increased construction costs and an increased schedule for construction and licensing.
(2) Identify credible secondary or concurrent hazards. These are referred to below as correlated hazards. Examples of correlated hazards include seismically induced floods or concurrent high wind and heavy rain.
(3) Define each of the hazard groups further to include specific hazard attributes that may impact or influence the effect of the hazard on the plant design.
(4) Identify the plant design relevant to each hazard group and evaluate the design for each hazard.
(5) Establish qualitative screening criteria. Example screening criteria are included in Refs [22, 32]. The screening criteria for deterministic treatment of a hazard may be different from that for the PSA, depending on the country-specific requirements for the plant. For example, quantitative screening may allow a beyond design basis event hazard to be screened if a sufficient margin is demonstrated for either the hazard magnitude (e.g. flood height) or plant/SSCs capacity. However, this may not result in

the hazard screening in the PSA. This potential mismatch would need to be addressed in any risk-informed process of optimization of safety measures for protection.

(6) In the case of qualitative screening, a hazard group is screened for further evaluation if all of the hazards within the group as well as potentially correlated hazards meet the qualitative screening criteria.

(7) Hazards groups that do not screen qualitatively are analysed further including the development of hazards intensity/severity frequency curves.

(8) Establish quantitative screening criteria. Example screening criteria are included in Refs [22, 32].

(9) For quantitative screening, screen a hazard group from further evaluation when all of the hazards within the group as well as potentially correlated hazards meet the quantitative screening criteria. Quantitative screening may need to consider the impact on multiple reactors (e.g. multi-unit risk) for sites with multiple reactors.

(10) Quantitative screening may be iterated further if either the grouping or treatment of correlated hazards is conservatively treated. This may include re-grouping of hazards or the development of hazard-specific correlation frequencies (e.g. the specific frequency for a high-wind and rain event exceeding specific magnitudes).

(11) Identify and document the remaining hazard groups that are not screened, including correlated hazards. These hazard groups are analysed in detail as discussed below but may be addressed through the process of optimization of safety measures discussed in Sections 4 and 5.

4.2.1.2. PSA Considerations for a Nuclear Installation in the Design Phase

For a nuclear installation where the site has not been selected, hazard PSA is performed in several phases:

(1) Identify hazards and hazard groups that may be possible for any potential site. This can be difficult, since each site will have a different set of potential hazards. As such, the list of potential hazards for generic sites of nuclear installation may be extensive:
   - The list developed is generally gathered from industry guidelines, publications and past studies, and supplemented by requirements and recommendations from authorities in the country of origin;
   - Grouping of hazards to simplify the analysis can occur at this stage or later following screening.

(2) The identified hazards are then qualitatively screened, based on the general nuclear installation design and capability. Screening criteria are discussed in Refs [22, 32]. Both qualitative and quantitative screening need to consider combinations of hazards, either concurrent or secondary.

(3) Develop the hazard frequency of occurrence or exceedance and the associated parameters (e.g. loading, magnitude) for quantitative screening. For a nuclear installation without a site selected, the hazard frequencies are based on bounding estimates for the potential sites.

(4) Once the hazard frequencies are estimated, the impact of each hazard is assessed by performing conservative estimates of each applicable risk metric such as CDF. Conservative estimates include overall conservative assessment of the SSCs impacted by each hazard, and the failure modes that occur including the hazard-induced initiating event. Sites with multiple reactors or modules need to consider the potential for a multi-unit event when measuring against any risk metric.

(5) Perform the detailed analysis for hazards not screened. Detailed analysis begins with the assessment of hazard-induced failure probabilities as a function of hazard intensities (i.e. fragilities) for each relevant failure mechanism modelled in the PSA:

- Fragility refinement can be performed in stages, with generic fragilities applied to non-risk-significant fragilities, and more detailed and realistic fragilities applied for risk-significant fragilities.
- Detailed fragility assessments may not be needed if the hazard is low risk overall. The assessment can include the use of generic or conservative fragilities for modelled SSCs failure modes, as long as the overall risk estimate is conservative [31, 32].

(6) The above fragilities, when combined with a more realistic hazard frequency estimate, are used to calculate the hazard-induced risk more accurately.

(7) Refinement of the hazard frequencies (hazard curves) can occur, although for a nuclear installation without a site, the estimate would still be bounding.
- If for example, the hazard curves are analysed in a discrete manner, the bins used can be refined to match more closely the calculated SSCs fragilities.
- Additionally, if a hazard risk is too high during the design phase, design requirements may be placed on the design that would limit the sites where the design is placed, resulting in a revised hazard curve. For example, if the seismic risk is calculated to be higher than desired for all potential sites, a design requirement may be imposed to limit the sites selected to moderate or low hazard sites.

For a site-specific nuclear installation, the above steps would include additional steps for refinement of the hazard curves. The development of detailed hazard curves for a site deterministic analysis or PSA can be a costly effort, with the level of detail proportional to the estimated radiological risk. This is an area of potential optimization of safety measures which is discussed in Section 6, where a site can potentially rely on more generic/bounding or less detailed hazard frequency curves for lower risk hazards.

4.2.1.3. Hazards PSA

Unscreened hazard groups are characterized further. The level of detail for each hazard group depends on both the hazard group (e.g. some hazards are more complicated to analyse such as seismic or tornado missiles) and the analysed risk; with risk-significant hazards need more detailed analysis in comparison. The following steps generally outline the hazards PSA process, although specific steps may be needed for many of the hazards:

(1) The hazards and hazards groups are further defined in terms of parameters and characteristics that support the evaluation of the nuclear installation impact and response such as SSCs impact. This definition includes all ranges of possible hazard severities including both lower end and higher end severities (to the physical limit of the hazard):
- Grouped events are bound by the worst case impacts within the group. Re-grouping can be performed if this grouping becomes a risk-significant impact.
- The definition of correlated hazards is also further refined in terms of nuclear installation impact and response. When considering multiple correlated hazards, the definition includes the parameters and characteristics for each potential combination of hazard as well as the independent hazard occurrence. For example, high wind and heavy rain may occur both separately and together, which would have unique frequencies.

(2) Develop the hazard frequency curves from hazards analyses

(3) Evaluate the nuclear installation capability using SSCs fragilities for each hazard and systems modelling. This step as well as the quantification step discussed below are

hazard-specific and include hazard specific steps not covered in this publication. For example, in fire PSA; the analysis includes fire growth and damage models and other fire-specific analysis steps. The general steps for evaluating the nuclear installation capability for each hazard includes:

- Define the nuclear installation safe shutdown capability for each hazard.
- Develop an SSCs list that supports the shutdown capability. For PSA, this may include all possible approaches to perform the nuclear installation safe shutdown capability for higher risk hazards.
- Determine the protection features for each SSCs, such as walls, systems or other nuclear installation features installed to protect the safe shutdown function.
- Compute the fragilities of each SSCs and protection feature for the hazard. Initial fragility may be assigned based on conservative estimates, such as generic/bounding fragilities. Risk-significant SSCs fragilities would be analysed in more detail as the hazard analysis is completed.
- In the design phase, fragilities are typically conservative especially prior to detailed design being complete; although the capability of the nuclear installation and specific SSCs is an area of potential optimization of safety measures discussed in Sections 4 and 5.

(4) Quantify each hazard group over the range of hazard severities. This may include dividing the hazard severity into frequency bins, with higher risk hazards divided into a larger number of bins. Refinement of the quantification is typical started with an initial conservative treatment, followed by further refinement of both the hazard curve and bins as well as more detailed assessment of the nuclear installation capability.

(5) The hazard analysis is documented for each hazard. Where needed; iteration of the hazard analysis may be performed using the process of optimization of safety measures for protection discussed below or in refinement of the analysis.

As discussed in the approach above, iteration of the hazard analysis maty be performed either in the design phase or as design details are developed (for a nuclear installation in the design process). This hazard analysis iteration can occur as a part of the process of optimization of safety measures discussed in Section 5. Risk or performance targets are needed for the installation/site to support the hazard analysis, or the process of optimization of safety measures. This is discussed in Section 6.

4.2.1.4. Determination of Risk-Significant Hazards

The risk significance of a hazard or SSCs can be measured using PSA risk importance measures such as Fussell-Vesely (F-V) and risk achievement worth importance measures. As discussed in Ref. [22], a risk-significant hazard (for existing nuclear installations) is a hazard contributing greater than 1% to the overall nuclear installation risk (risk is determined by the applicable risk metric as discussed in 4.2.1.2). Conservatively estimated hazards, contributing greater than 1% of risk may ultimately be screened if the estimated final risk using more detailed and realistic analyses shows a contribution of less than 1% of the total risk. Screening criteria need to be established such that a potentially significant hazard is not screened.

Risk significance for SSC for existing installations can be defined either based on the overall nuclear installation risk or on a hazard-specific basis (e.g. contributing a specific percentage for the analysed hazard). A risk-significant SSC is generally defined as contributing greater than 0.5% of the risk or having a risk achievement worth greater than 2.

Risk significance may be difficult to estimate for hazards where the hazard analysis approach or the hazard frequency data is not fully developed. This is the case for rare events, such as volcanic hazards for example, or combinations of hazards where the frequency versus magnitude is difficult to estimate – including the frequency for various combinations of hazard magnitudes. This can also affect estimations of more realistically estimated hazards since the risk significance of an individual hazard is based on the contribution of the hazard to the overall risk. If a conservatively analysed hazard or set of hazards is skewing the overall results for site risk, this need to be considered in estimating risk significance. One option is to consider risk significance of a hazard based on contribution for the total nuclear installation risk with the contributions from conservatively analysed hazards removed. Alternate approaches are also possible.

For advanced plants, Ref. [33] proposes alternate criteria for plants with an overall lower risk than the existing fleet. The proposed approach is to apply absolute risk measures versus relative, with the absolute risk metric based on the plant QHOs. Using this approach, a risk-significant hazard would be a hazard contributing greater than 1% compared to any applicable QHO, such as offsite dose, latent cancer fatalities, etc. If the plant risk for an advanced plant is well below the QHOs, the use of absolute risk metrics provides more opportunity for risk optimization of safety measures.

4.2.1.5. Identification of Significant Risk Contributors

Risk significance is defined in Ref. [22] and can be determined for each hazard and for equipment relied on to respond to a hazard or set of hazards. During the initial analysis involving the use of either bounding hazard curves or conservative assumptions related to plant response, the determination of risk significance can be used to help focus analysis refinement or even design enhancements or changes. Although this is generally applied to identify where either additional analysis is needed or where plant enhancements may be considered, in some cases, the analysis or application of design requirements may be reduced. During the site specific phase, since the design is generally established, the design changes may be limited to a reduction of application of requirements, which could result in a reduction in the regulatory costs or commitments. This would be the case, for example, if a plant design which is robustly designed for a specific hazard but is built on a site where this hazard is either not applicable or low risk.

Risk significance is one of the key inputs to the optimization of safety measures for protection. Risk significance can be calculated for equipment or plant features for individual hazards, or for the integrated plant risk. The risk significance calculated for an individual hazard can be used to identify equipment or plant features where optimization of safety measures could have the greatest effect. The risk significance for hazards calculated for the integrated plant risk can be used to identify hazards that can either be potentially screened, have reduced hazards qualification requirements (as discussed in Section 3.3), or where additional optimization of safety measures can be implemented to reduce the single hazard risk.

4.2.2. Risk and performance targets

The established PSA risk metrics would determine whether it is necessary to perform either a Level 1, Level 2 or Level 3 PSA respectively, as discussed in Refs [23, 31]. Safety goals or criteria may not specify which hazards or plant operational states have to be addressed. In order to use the PSA results for the assessment of risk against existing safety goals or criteria, a full scope PSA involving a comprehensive list of initiating events and hazards and all plant operational modes needs to be performed unless the 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 modes that are not in the model does not threaten compliance with the safety goals or criteria. As with existing LWRs, when the PSA scope includes bounding or conservative approaches for selected hazards or scenarios, the results may distort the risk profile and impact the calculation of risk importance measures use for PSA applications.

As discussed in Section 6, any quantification of the PSA results against a site risk metric for a site containing multiple units or modules may need to consider the potential for a multi-unit release from any analysed external hazard.

4.2.3. Risk management

In general terms risk management is a process for obtaining a robust and balanced defence in depth using a risk informed and performance-based approach aimed to meet the safety objectives. The process is iterative and results in an improved design based on feedback from safety assessment of the design.

Steps in the process includes:

(a) Develop the design observing the applicable design safety requirements and associated guidelines;
(b) Perform the safety evaluation of the design (at a given design phase using deterministic and probabilistic methods);
(c) Use the results of the PSA discussed in Section 4.2.1 to develop risk insights for each hazard or group of hazards, including insights related to the relative importance of each potential hazards sequence;
(d) Perform risk informed judgement of the performance of the design related to the intent of the safety requirements and to the level of adequacy of the defence in depth attributes;
(e) Optimize/improve performance targets associated to groups of design provisions;
(f) Consider other aspects (principles) of IRIDM, as presented in Ref. [21] (see Section 4.1);
(g) Develop risk management actions including:
    (i) Identifying techniques, tools, or strategies to manage the risk. Three categories of risk management tools are; risk reduction, risk retention, and risk transfer. In many cases, a combination of tools, techniques, and strategies will be used, rather than a single approach.
    (ii) Implementing the chosen techniques or strategies.
    (iii) Monitoring the effectiveness of solutions, providing feedback so that risk analysis is always updated as the operating environment changes.
(h) Provide feedback for improvement of the defence in depth.

Improved design can be obtained by setting risk informed performance targets to the groups of design provisions and modifying the design to reach these performance targets for enhancing the independence of the defence in depth barriers, for getting an improved distribution of the safety margins and for reduction of the risks associated to the loss of multiple defence in depth barriers.

5. OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION AGAINST HAZARDS PRIOR TO SITE SELECTION FOR ADVANCED REACTORS

Prior to site selection, the hazard analysis is performed using generic/bounding hazard parameters. Such generic hazard parameters are used to develop the nuclear installation design. Selection of bounding hazard parameters is mainly a decision made by the designer that includes potential regions and sites where the installation can be built with minimum design changes due to specific site conditions. Once a specific site is selected and characterized the safety of the nuclear installation on that specific site can be demonstrated by a simple comparison between site specific parameters and those used in the design.

The PSA is typically used for providing feedback to the designer and to help to improve the design during the design development.

5.1. BOUNDING SITE EVALUATION OF HAZARDS

Prior to performing detailed analysis, the hazard investigation and data collection is performed. Most of the information collected (e.g. intrinsic hazards present, thermal power of the reactor, etc.) are applicable to the nuclear installation design either prior to site selection or after the site is selected. However, some of the information, such as the site specific installation factors, environmental factors such as weather conditions or demographics, etc. will be finalized once the site is selected. For analysis prior to site selection, this information is either assumed or based on bounding site characterization. Specific steps performed in preparation for the process of optimization of safety measures include:

(1) Depending on the installation type and design characteristics; define each hazard, the hazard category (e.g. Small, Medium and High) of the installation and associated performance goal(s) to be considered:
    (a) This is discussed further in the section below. This involves the identification of bounding hazard characteristics that bounds all potential sites where the design may be built. As discussed below, limitations may be needed on the bounding hazard characteristics in order to ensure the hazard is not risk significant or the design against a specific hazard is not unduly burdensome.

Once the bounding site is defined and characterized, screening of the hazards can be performed as described in Section 5.3 and quantitative hazard evaluations are performed as described in Section 5.4.

5.2. HAZARD SCREENING PROCESS PRIOR TO SITE SELECTION

The screening of external hazards includes both qualitative and quantitative screening. The factors used to qualitatively screen the hazard in the PSA can also be applied to the deterministic evaluation. The quantitative screening process in the PSA can likely be applied to the deterministic approach but would depend on the regulatory requirements of the country where the nuclear installation is being built, and the reason for screening the hazard.

When a hazard is screened out from the PSA, this indicates that the nuclear installation requirements are likely reduced or eliminated. However, the reason for the screening of the hazard can impact this determination, as well as the consideration of uncertainty including consideration for a potential cliff-edge effect. If, for example, the hazard has been screened out due to a low frequency – there may be no protection of safety measures needed in a risk-informed approach. However, if the frequency of the hazard is above the design basis event

(DBE) level criteria (e.g. 1E-04/year), and the hazard is screened out due to nuclear installation's capability, then the basis for the nuclear installation's capability including margins are analysed for potential protection of safety measures. An example of this would be high or low temperature at a site, where the frequency of exceeding design limits may be with the DBE criteria, but low risk impact may result in screening from the PSA using quantitative screening criteria. This is further discussed in Section 5.4.

For a nuclear installation where the site is not yet selected, hazard screening is performed using the steps in 4.2.1.1 as supplemented with the following:

(1) The initial assessment involves the identification of hazard groups. This can be difficult, since each site will have a different set of potential hazards. As such, the list of potential hazards for a nuclear installation where the site is not yet selected may be be extensive:
   - The list developed is generally gathered from industry guidelines, publications and past studies;
   - Grouping of hazards to simplify the analysis can occur at this stage or later following screening.

(2) Screening criteria are discussed in Refs [22, 32].
   - Both qualitative and quantitative screening need to consider combinations of hazards, either concurrent or secondary;
   - Optimization of safety measures may be possible at this stage by making changes in the design such as additional capability that results in a hazard being screened, as discussed in the Section 4 introduction;
   - Additionally, optimization of safety measures for any potential design changes may consider the optimization of safety measures against multiplied hazards, such as design changes to remove onsite emergency power requirements for ensuring nuclear installation safe shutdown.

(3) Quantitative screening initially involves the development of the hazard frequency of occurrence or exceedance and the associated parameters (e.g. loading, magnitude).

(4) For hazards that do not quantitatively screen, these are analysed in detail as discussed in Section 5.3.

As discussed in Step 2 above, optimization of safety measures during the screening process may involve potential design changes which can improve nuclear installation robustness for one or more hazards. Optimization of safety measures at this stage may result in the elimination of requirements for both the site and the design related to the potential hazard. One additional step involves refinement of the bounding hazard curve and can also be performed during the hazard screening process. This step may potentially eliminate design requirements related to the potential hazard. However, this step may also result in requirements on the site selection process (e.g. place limitations on the sites where the nuclear installation can be placed). This step, which can be performed at any stage of in the design development process, is discussed further in Section 5.4.

5.3. HAZARD PSA FOR A BOUNDING OR GENERIC SITE

Hazard analysis during the design phase can vary, depending on the defined bounding characteristics. As discussed in Section 5.1, this hazard curve would generally be based on a bounding site. Optimization of safety measures of this bounding hazard curve is discussed further below, as part of optimization of safety measures for the design improvements applied during quantification.

Hazard analysis is performed as follows:

(1) Refine combination of hazards such as high winds and external flooding. For example, separate hazard frequencies may be developed for high winds, external flooding and the combination of both hazards;
(2) The impact of each hazard is evaluated by performing conservative estimates of each applicable risk metric such as CDF. Conservative estimates include conservative assessment of the SSCs impacted by each hazard, and the failure modes that occur including the hazard-induced initiating event:
    - Sites with multiple reactors or modules would need to consider the potential for a multi-unit event when measuring against any risk metric.
(3) Detailed analysis begins with the assessment of hazard-induced failure probabilities as a function of hazard intensities (viz. fragilities) for each relevant failure mechanism modelled in the PSA:
    - Fragility refinement can be performed in stages, with generic fragilities applied to non-risk-significant fragilities, and more detailed and realistic fragilities applied for risk-significant fragilities;
    - Detailed fragility assessments may not be needed if the hazard is low risk overall. The assessment can include the use of generic or conservative fragilities for modelled SSCs failure modes, as long as the overall risk estimate is conservative [31, 32].
(4) The above fragilities, when combined with hazard frequency estimates, are used to calculate the hazard-induced risk:
    - If sufficient design details are available to location-specific hazard impacts, such as incorporation of in-structure response spectrum for seismic hazards, this need to be included during this step.
(5) Refinement of the hazard frequencies (hazard curves) can occur, although for a nuclear installation without a site, the estimate would still be bounding.
    - If for example, the hazard curves are analysed in a discrete manner, the bins used can be refined to more closely match the calculated SSCs fragilities;
    - Additionally, if a hazard risk is too high during the design phase, design requirements may be placed on the design that would limit the sites where the design is placed, resulting in a revised hazard curve. For example, if the seismic risk is calculated to be higher than desired for all potential sites, a design requirement may be imposed to limit the sites selected to moderate or low hazard sites.

If this bounding hazard curve results in higher than desired risk results for the quantified hazard, optimization of safety measures may involve refining the design changes to improve nuclear installation robustness against the hazard or to modifying the hazard curve to remove conservatism. In some cases, this may result in requirements for the selected site or sites to be within the assumed hazard curve bounds. This interaction between the design robustness and the limitations placed on the site and site requirements is possible for many hazards, such as seismic hazards, high winds, external flooding, etc.

5.4. OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION DURING DESIGN PROCESS AGAINST SPECIFIC HAZARDS

During the design process prior to site selection, the designer has an opportunity to optimize safety measures of protection in relation to the design against internal and external hazards. This includes the modification of the nuclear installation design to better respond to one or more hazards, and/or to achieve a balanced distribution of the margins against specific hazards and/or to reduce risk contribution of the short accident sequences involving failure of the safety passive systems. An example of optimization of safety measures during the design phase is provided in Annex I.

A design decision could be to develop a very robust design of the SSCs related to inherent passive safety functions in a way that their contribution to the nuclear installation risk related to external and internal hazards remain very low (not significant). This early design decision may show benefits later on as described in Section 6. As noted in Section 2.3, the process of optimization of safety measures needs to include the evaluation against all hazards and initiating events to ensure overall safety goals are attained.

In the case of a nuclear installation that is proposed to be built at a large number of sites, developing a robust design against bounding site characteristics not only results in lower overall risk, but also ensures the reactor design can be constructed on most sites proposed. In most cases, the design would provide significant margins when the site specific hazard is much lower than the bounding site characteristics applied to the generic design. The advantages during the site licensing process may be significant, as discussed further in Sections 6 and 7.

In some cases, the reduction in risk may be as a result of risk management actions, such as the general areas discussed in Section 4.2.3. If risk management actions are implemented, monitoring or other operational requirements may need to be implemented.

5.5. RISK SIGNIFICANCE OF HAZARDS PRIOR TO SITE SELECTION

Risk-significant items are identified, prior to site selection, based on risk ranking as described in Section 4.2.1.5 using a design phase PSA and as well as generic/bounding hazard parameters. Risk ranking is performed using importance measures, and sensitivity analysis results aimed to show the SSCs significance to risk associated to analysed hazards. The risk significance associated with SSCs supplemented by other considerations related to the defence in depth robustness in relation to analysed hazards provide the basis for making risk-informed decisions supporting the optimization of safety measures for protection during the design stage.

5.5.1. Optimization of safety measures for distribution of margins against external hazards

The purpose of the process of optimization of safety measures for hazard assessment is to enhance the design robustness against external and internal hazards in order to:

- Provide sufficient margins to avoid cliff edge effects;
- Provide sufficient margins to prevent large early releases;
- To reduce risk contribution of the short accident sequences;
- Generally, to minimize hazards contribution to the risk;
- Balance the efforts for application of design requirements based on the risk contribution for a given hazard.

Although the application of this process may be unique to each hazard and hazard group, Fig. 1 provides an outline of the general process for each step of potential optimization of safety measures for protection.

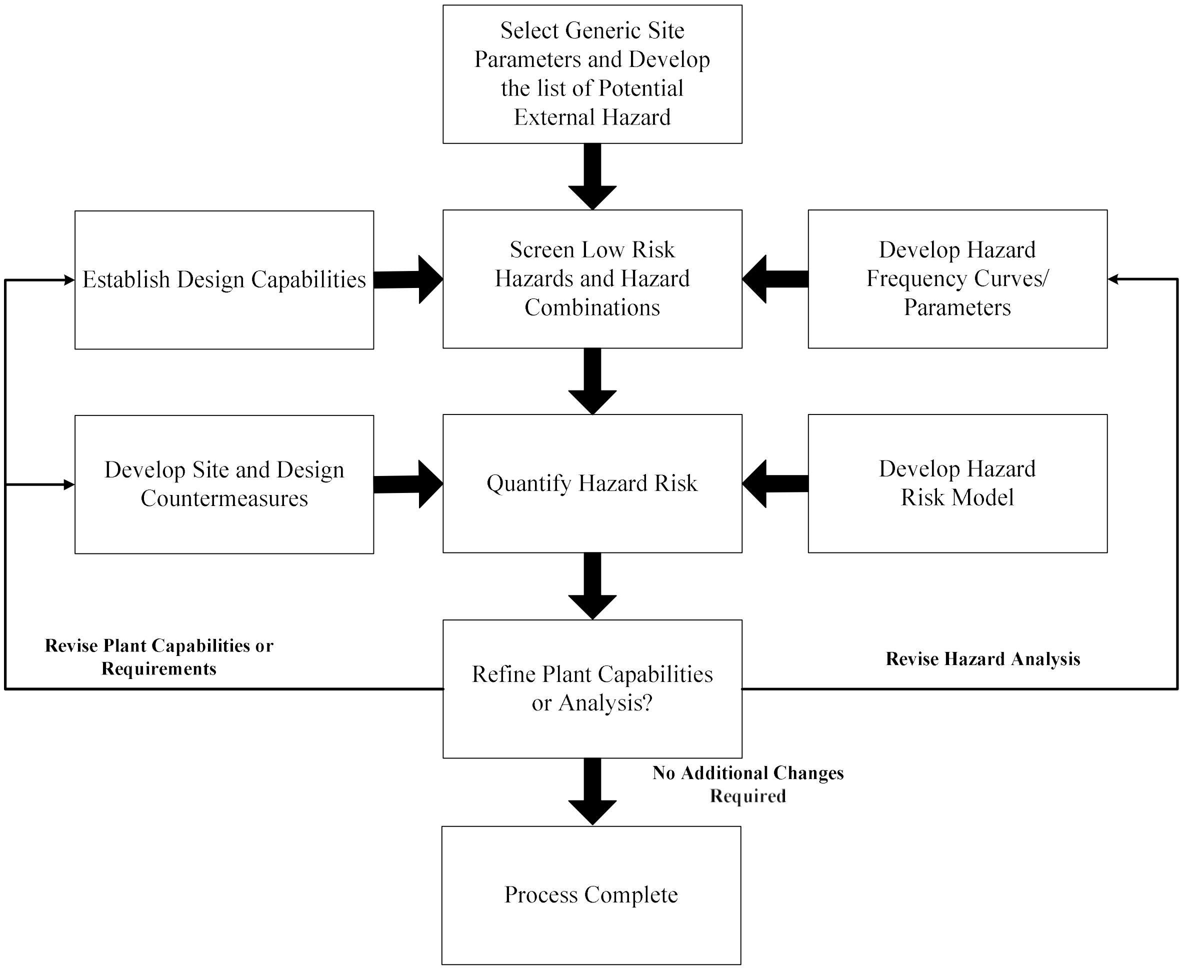

FIG.1. General process of optimization of safety measures in relation to hazards prior to Site Selection.

6. OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION FOR A SITE SPECIFIC APPLICATION

A site specific application for an advanced reactor may involve either selecting a nuclear installation for a proposed or existing site, or application of a pre-selected design to one or more sites within a country. In both cases, this involves the comparison of the hazards assumed in the analysis for both analysed and screened hazards to the site specific hazards. Section 6.1 provides information for either an operating organisation or regulatory body when considering multiple reactors for a selected site, including the specific considerations for the site hazards in comparison to the proposed designs. Sections 6.2–6.6 provide information on analysing a specific design against a known or selected site.

6.1. SELECTION OF A REACTOR DESIGN AND A SUITABLE SITE

When selecting a specific reactor design for a selected or proposed site, the selection process can be complicated, involving factors beyond the issues and processes discussed in this publication. The overall selection process needs to consider the site characteristics, including possible external hazards, by comparing with those used for the proposed design.

The site selection process is described in Refs [10, 34]. It is advisable to already have one or more reactor technologies proposed for consideration when performing site selection.

The objective of site selection is to establish and follow a systematic process for site survey and site selection for applying to one or more sites meeting the site suitability criteria for the reactor technologies considered for construction at that site(s) and meeting the applicable safety requirements. This is an iterative process aimed at grading the effort and data collection supporting the site-installation suitability assessment.

Requirement 4 of SSR-1 [10] states that "**The suitability of the site shall be assessed at an early stage of the site evaluation and shall be confirmed for the lifetime of the planned nuclear installation.**"

Paragraph 4.6 of SSR-1 [10] states:

> "In the assessment of the suitability of a site for a selected nuclear installation the following aspects shall be addressed at an early stage of the site evaluation:
>
> (a) The effects of natural and human-induced external events occurring in the region that might affect the site
> (b) The characteristics of the site and its environment that could influence the transfer of radioactive material released from the nuclear installation to people and to the environment;
> (c) The population density, population distribution and other characteristics of the external zone, in so far as these could affect the feasibility of planning effective emergency response actions, and the need to evaluate the risk to individuals and to the population."

Para 4.7 of SSR-1 [10] states:

> "The site shall be deemed unsuitable for a proposed nuclear installation if one or more of the three aspects listed above indicates that the site is unacceptable, and the deficiencies cannot be compensated for by means of a combination of measures for

site protection, design features of the nuclear installation and administrative procedures."

For practical reasons, two sets of information are needed:

(a) Information related to the reactor technology considered;
(b) Site characteristics, including historical data.

Paragraph 4.8 of SSR-1 [10] states:

> "Site suitability shall be assessed based on relevant current data and methodologies. If relevant, conservative criteria shall be developed in relation to site specific accident scenarios, and the consistency of such criteria with the overall site suitability shall be demonstrated."

A decision regarding the suitability of the site needs to be based on the characteristics of the nuclear installation, including planned operations at the site, the amount and nature of potential radioactive releases and their impact on people and the environment.

Detailed suitability criteria are based on a comparison between the generic bounding site parameters used for design and the corresponding/relevant site specific parameters.

Example of information regarding nuclear installation parameters and site specific parameters are provided in Tables III–1 to III–3 of Annex III. After checking site suitability against exclusionary criteria (Table III–3) some key site factors are compared against applicable nuclear installation requirements for assessing site suitability. Site suitability conditions need to be confirmed later on when site evaluation information become available.

At the initial phase of the site evaluation, preliminary hazard estimates can be performed using available data and limited site specific survey information. The preliminary site specific hazard parameters can be compared to the bounding hazard parameters, to provide an estimated risk for the hazard. If, for example, the nuclear installation design analysis estimated a hazard to be a low risk contributor, and the site specific hazard curve was lower than the bounding hazard curve used in the analysis for design, then the conclusion that the hazard remains a low risk and a graded approach can be applied in the evaluation of the hazard parameters would be easily supported. However, if a specific hazard risk was estimated to be high during the design analysis, or if the site specific hazard curve was significantly higher than the bounding hazard curve used during design, then analysis steps to support the design selection may be more complicated. Similarly, if the hazard was screened for the nuclear installation design, either because the hazard was not applicable or due to a low hazard frequency, but the site specific hazard frequency is higher, additional evaluation steps will be needed.

Based on the preliminary hazard evaluation for a selected site and potential reactor designs, the preliminary safety assessment has to be used to:

(1) Verify whether the reactor will ensure the site risk estimates are acceptable for any QHOs or risk metrics, with margins.
(2) Evaluate each design against identified potentially risk-significant hazards to help determine:
    - If the potentially risk-significant hazards are likely to be risk-significant for each of the evaluated designs;

- If one of the designs clearly has a lower risk than other designs for evaluated external hazards. This evaluation may also need to consider risk from internal events to fully understand the likely overall nuclear installation risk.

(3) If the evaluated risk for all nuclear installations is similar, the assessment needs to consider uncertainty including the potential cliff-edge effects for each design. This would include whether each design would be likely to experience a direct core damage or release given a hazard exceeds a specific severity.

6.2. SITE SPECIFIC HAZARDS SCREENING

Requirement 6 of SSR-1 [10] states:

"**Potential external hazards associated with natural phenomena, human induced events and human activities that could affect the region shall be identified through a screening process.**"

Paragraph 4.16 of SSR-1 [10] states:

> "The process and associated criteria used in the screening of site specific hazards shall comply with the safety objective for site evaluation and shall be properly justified and documented."

Paragraph 4.17 of SSR-1 [10] states:

> "The scope of evaluation of external events in the screening process shall cover the full ranges of severity and frequency of occurrence relevant for the design and the safety assessment of the nuclear installation, including events of high severity but low probability that could contribute to the overall risk."

Paragraph 4.18 of SSR-1 [10] states:

> "An event might be screened out because it is enveloped by a set of events. However, it shall be ensured that all potential effects of the screened-out event are bounded by this set of events."

Paragraph 4.19 of SSR-1 [10] states:

> "External hazards that are not excluded by the screening process shall be evaluated and then used in establishing the site specific design parameters and in the re-evaluation of the site, in accordance with the significance of these hazards to the safety of the nuclear installation".

As discussed in Section 3 of this publication, many of the hazard groups including the combination of hazards can be screened prior to the site being selected. After the site is selected the site specific hazard screening needs to be conducted. The objective of the site specific screening is to narrow down the number of site-specific hazards and their potential combinations that need to be considered for design and safety assessment evaluation. Unscreened hazards are analysed and are potential candidates for a more detailed hazard assessment. The general steps and the process of optimization of safety measures for this refinement are as follows:

(1) Identify the possible site specific hazards and hazard groups. This identification has to begin with the hazards identified in the design process, prior to the site being selected. Each site will have a different set of potential hazards. As such, the list of potential hazards for a specific site may be extensive, although generally smaller than the hazards list developed prior to the site selection.

(2) Review the identified hazards for the selected site or sites. Any new hazards identified for the site are added to the potential hazards list for the design:
   - In general, application of a design to a specific site ought not to identify new hazards if the initial hazards review is comprehensively performed.
   - More likely, the initial screening or grouping of hazards may need to be refined, based on site specific details.

(3) Screen the identified hazards, based on the general nuclear installation design and capability. Screening criteria are discussed in Refs [22, 32].
   - Both qualitative and quantitative screening would need to consider combinations of hazards, either concurrent or secondary.
   - Optimization of safety measures may still be possible at this stage by making changes in the design such as additional capability that result in a hazard being screened, as discussed in the Section 4. However, if the nuclear installation design is generally complete, as with a nuclear installation being constructed on multiple sites, the ability to change the design may be more difficult other than site specific features such as the refinement of mobile equipment capabilities.
   - Additionally, optimization of safety measures for any potential design changes may consider the optimization of safety measures against the multiplied hazards, such changes may include to remove on-site emergency power hazards qualification requirements for ensuring nuclear installation safe shutdown.

(4) Quantitative screening initially involves the development of the hazard frequency of occurrence or exceedance and the associated parameters (e.g. loading, magnitude). For a nuclear installation with a site selected, the hazard frequencies are based on preliminary site specific estimates:
   - Once the hazard frequencies are estimated, the impact of each hazard is evaluated by performing conservative estimates of each applicable risk metric such as CDF/LERF. Conservative estimates include overall conservative assessment of the SSCs impacted by each hazard, and the failure modes that occur including the hazard-induced initiating event.
   - Sites with multiple reactors or modules would need to consider the potential for a multi-unit event when measuring against any risk metric.

(5) For hazards that cannot be screened out quantitatively, these are analysed as discussed in Section 6.3.

6.3. DATA COLLECTION SUPPORTING SITE EVALUATION

A graded approach for site evaluation process generally has to be applied considering level of details, and amount of data collection and analyses needed at each phase of this process. All the relevant data that may be used for safety evaluation of nuclear installation is generally not available in the site selection phase. As additional data is collected and analysed for a candidate site, the ability to optimize safety measures for the particular design increases.

Factors considering site installation interactions are to be identified and considered in the application of the graded approach. Factors related to site installation interactions to be considered in setting up the graded approach include:

- The characteristics of engineered safety features for the prevention of accidents and for mitigation of the consequences of accidents, including the need for active safety systems and/or operator actions for the prevention or mitigation of accidents;
- Margins against external hazards;

- The site characteristics that may challenge the safety and mitigation functions of the installation;
- Source term, and the frequency of the radioactive release (if calculated);
- The site characteristics that may influence dispersion of radioactive material;
- Estimation of the risk contribution of the site specific hazards;
- The level of detail needed in the evaluation of a site for a nuclear installation shall be commensurate with the risk associated with the nuclear installation.

Prior to performing detailed analysis, data collection and site investigation are performed. Most of the information collected (e.g. intrinsic hazards associated with the physical process of the nuclear installation, thermal power of the reactor, etc.) are applicable to the nuclear installation design either prior to site selection or after the site is selected. However, some of the information, such as the site specific installation factors, environmental factors such as weather conditions or demographics, etc. will be refined once the site is selected.

Application of the graded approach for site specific data collection typically considers the following aspects:

- Identification of the sources of needed data;
- The quantity and quality of the available data (collected during site selection phase);
- The need for additional data collection (limited in space and time);
- Conservative assumptions and increased uncertainties complementing the use of reduced data analysis for hazard assessment;
- Design margins against site specific hazards (using preliminary hazards estimates);
- Site specific external hazard estimated contribution to the risk;
- Potential limitation of data in (in space and time);
- Impact of increased uncertainties due to limited data used in preliminary hazards estimates.

6.4. SITE SPECIFIC HAZARD ANALYSIS

Preliminary hazard estimates can be used to assess the risk associated with that hazard using external hazard PSA. If the hazard risk contribution is lower than a threshold, then it can be concluded that there is no need to refine the hazard estimates, and preliminary results could be used to finalize the design. In this way, site evaluation time, effort, and resources may be optimized. As noted in Section 3.4, for high hazards sites, the evaluation for a specific hazard may need to be fully characterized to show the full benefit of optimization of safety measures and the risk-informed process, especially for risk-significant hazards.

At the facility level for screened-in hazards, a RIPB approach can be utilized to obtain a better distribution of the margins for each considered hazard.

The safety of the installation of a specific site can be demonstrated if the hazard parameters used for design envelop the site specific parameters. Site characterization, involving development of the site specific hazard parameters, is a resource intensive activity and may require significant time (typically 3–4 years for an NPP site).

A graded approach for hazard characterization can be applied, considering the completeness and quality of the information, data, and site investigation level of detail. One general approach to optimize the safety measures for the hazard characterization is as follows:

- Select the methodology to be used for the preliminary site specific hazard assessment;
- Review the existing available information and data and the need for new investigations to collect more data;
- Define the scope of the site investigations considering application of the graded approach;
- Develop the hazard assessment model and develop preliminary evaluation of the hazard parameters (frequency and severity parameters for scenario base hazard data and frequency severity curves for hazards characterized by hazard curves, considering associated uncertainties);
- Estimate the hazard impact and its risk contribution;
- Refine the external hazards parameters if needed;
- Develop the final hazard results;
- Document the results including definition of the limitations, restrictions, and assumptions of the analysis.

When applying a design to a specific site, assuming the site is bounded by the site characteristics assumed in the design process and meets other regulatory considerations, the site specific hazard analysis can benefit from the process of optimization of safety measures discussed in Sections 6.4 and 6.5. This includes reduced effort in the site specific analysis discussed above, including data collection and hazard impact analysis, which can utilize bounding estimates for each.

The general approach listed above is discussed further and an example of a graded approach to PSHA is presented in Annex II.

The following major elements of PSHA are presented in Annex II:

- Review the existing seismotectonic database;
- Use conservative seismic sources parameters, selection of Ground Motion Prediction Equations (GMPEs) and site response parameters;
- Perform preliminary PSHA and sensitivity analysis to identify the more important PSHA model elements (e.g. seismic sources, seismic source parameters, etc.);
- Decide additional investigations and data collection for the important PSHA model parameters (limited space i.e. reduce the area of investigations to a radius from site such that at least 90% of the hazard for all structural frequencies and all annual frequencies of exceedance are captured, limit data collection supporting seismic source characterization and use conservative estimates of the seismic sources parameters, etc.);
- Use Backbone model for the GMPEs to capture full uncertainty of the selected GMPEs;
- Develop PSHA results and estimate the hazard contribution to the nuclear installation risk. If the risk contribution is not significant, no detailed hazards assessment needs to be conducted, otherwise updated/refine the preliminary hazard assessment.

The example of seismic hazard assessment and a comparison of the simplified approach with SSHAC Level 3 results is provided in Annex II. The SSHAC Level 3 process is a formal, structured procedure for developing Seismic Source Characterizations and Ground Motion Characterizations (GMC) and has been identified in Ref. [35] as an acceptable method for use in performing PSHA.

6.5. OPTIMIZATION OF SAFETY MEASURES FOR EQUIPMENT QUALIFICATION

For a selected site and with a detailed design completed, additional optimization of safety measures based on a risk-informed approach can be applied to explore the relaxation (or strengthening) of hazard qualification requirements (without any significant design change).

Additionally, the overall nuclear installation risk can be estimated to ensure that the nuclear installation meets the risk and performance goals for the site specific conditions. The attributes supporting the risk-informed process is discussed in Section 3, while the process of optimization of safety measures performed prior to site selection and for a specific site is discussed in Sections 4 and 5.

When a specific site was selected, and a nuclear installation is being constructed on multiple sites, the ability to change the design may be more difficult other than site specific features such as the refinement of mobile equipment capabilities.

However, the risk significance in relation to site specific external hazards may help to reduce the generic based qualification requirements against external hazards (mainly in the equipment specification data sheet that is sent to the equipment suppliers) without involving any significant design changes.

This type of risk-informed optimization of safety measures may reduce the cost and time associated with procurement of equipment components without making changes to the design.

In case if qualitative analysis and quantitative analysis for a hazard is not possible, the optimization of safety measures for the design for that hazard would be difficult.

6.5.1. Identification of significant risk contributors

Risk-significant items are identified based on risk ranking as described in Section 4.2.1.5, and on design phase PSA and site specific hazard parameters. Risk ranking is performed using importance measures, and sensitivity analysis results aimed to show the significance to risk associated to analysed hazards.

The risk significance associated with SSCs supplemented by other considerations related to the defence in depth robustness (including margins) in relation to analysed hazards provides the basis for making risk-informed decisions supporting the process of optimization of safety measures performed at the construction stage.

Risk significance is one of the key inputs to the optimization of safety measures for protection. Risk significance can be calculated for equipment or nuclear installation features for individual hazards, or for the integrated nuclear installation risk. The risk significance calculated for an individual hazard can be used to identify equipment or nuclear installation features where optimization of safety measures can have the greatest effect. The risk significance for hazards calculated for the integrated nuclear installation risk can be used to identify hazards that can either be potentially screened, have reduced hazards qualification requirements, or where additional optimization of safety measures can occur to reduce risk from the individual hazard.

6.5.2. Optimization of safety measures for qualification against external hazards

Based on site specific external hazards, the PSA for each hazard considered in the assessment identifies the risk contribution of items that have design provisions for qualification against external hazards (EH). Items qualification requirements associated to equipment data sheet are reviewed and items those with EH qualification requirements and with insignificant risk contribution are identified. It is expected that these items will include active mechanical and electrical equipment components. Below are next steps for equipment qualification against external hazards following the process of optimization of safety measures:

- EH fragility parameters corresponding to "No EH qualification conditions" are developed and their risk contribution is re-assessed.
- If items became a significant risk contribution as a result of fragility changes, no relaxation is possible.
- If items that remained non-significant risk contributors after changing fragilities parameters (to correspond to non-EH qualification), then the EH qualification requirements may be removed.

Removing EH qualification requirements from the equipment specification that are sent to an equipment supplier will reduce the cost of the items with insignificant design changes and may reduce the procurement time.

6.6. OPTIMIZATION OF SAFETY MEASURES FOR PROTECTION OF A SITE WITH MULTIPLE UNITS

The approach to MUPSA was revised after the accident at the Fukushima Daiichi NPP [27]. MUPSA is aimed to identify and analyse multi-unit risk contributors needed to support the site-wide risk management.

Shared safety systems can be both beneficial and disadvantageous during an accident. The MUPSA methodology can help countries to investigate and assess multi-unit site designs, covering both the positive and negative aspects of shared systems in a balanced way.

The MUPSA may provide a technical basis for maximizing the benefit of sharing systems and resources in the multi-unit context and minimizing the multi-unit accidents contribution (due to shared systems and resources) to the overall site risk. Moreover, MUPSA may support risk informed assessment of the adequacy of the defence in depth in a multi-unit context.

Ref. [27] provides a methodology for conducting MUPSA and assessing multi-unit contribution to the site risk. Ultimately, this methodology will help decision makers identify where to focus their resources within a multi-unit context to get the most benefit in terms of safety and cost.

7. REGULATORY FRAMEWORK AND USE OF GRADED APPROACH

The use of a graded approach for protection against external hazards described in the previous sections may be difficult when applying this approach on a specific national regulatory framework. Application may require modification or revision to the regulatory requirements or processes, or analysis supporting gradual application of the requirements – if allowed by the regulatory process.

Advanced reactors, especially NLWRs, may already need relief or exemptions from regulatory requirements that have been traditionally developed to support LWR reactors. For example, regulatory requirements for containment typically include consideration for water and steam releases, such as following a LOCA, which might not be a consideration for a NLWR such as a molten-salt reactor.

The following subsections provide an overview of the various regulatory frameworks, which implement a range of graded approaches, and discuss the benefits for the regulatory body, nuclear installation designer and operating organization of implementing optimized approach of safety measures for protection prior to an advanced reactor being built, especially in terms of protection against external hazards.

7.1. LICENSING APPROACHES FOR PROTECTION AGAINST EXTERNAL HAZARDS

Following subsections provide details on different approaches of licensing for protection against external hazards.

7.1.1. Deterministic Approach

The deterministic approach is mostly used by regulatory bodies and industry worldwide to provide confidence in a reactor design's defence in depth capabilities and is based on analysing postulated accident scenarios using approved conservative codes and criteria.

The conservatisms added to design limits, acceptance criteria, and safety margins are intended to manage the uncertainties associated with accidents, including possible "unknown unknowns," at the time a nuclear installation was designed.

Realistic conservatism means that decisions are informed by the real world of scientific knowledge, technological capabilities, and experience, in order to preserve appropriate and prudent safety margins, to regulate in a manner that corresponds to the actual risk presented and not to worst case assumptions. Being realistically conservative is in the best interest of public safety and the environment, by ensuring that the right balance between under-regulating — which puts the public safety and the licensees investment at risk — and over-regulating, which could divert resources from important safety issues while increasing costs to licensees and thus to consumers, without a matching safety or security benefit. Such an approach is also essential in order to understand the real margins to safety that exist in nuclear installations.

Safety margins are included in the analyses such that specific barriers are designed and constructed to ensure actual failures are not expected until key parameters well exceed the values assumed in the supporting engineering evaluations.

Important limitations of this traditional regulatory approach are that (1) significant accident scenarios may not be identified or addressed by the defined barriers and controls, (2) the stylized analyses and related barriers and controls may misdirect resources to address low-risk scenarios, and (3) excessive conservatism or the imposition of requirements that do not result in a proportional benefit to safety or only add minimally to safety beyond an already existing adequate level of safety can be contrary to an efficient and effective regulatory framework.

7.1.2. Risk-Informed Approach

Risk-informed approaches have been developed over the last several decades to supplement the traditional regulatory approach by doing a more methodical assessment of the risk triplet questions (what can go wrong, how likely it is, and what are the consequences).

A risk-informed approach to regulatory decision making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety.

The risk is the likelihood and consequences of an accident occurring in an installation or process. In risk assessment, the damage associated with a hazard is usually referred to as the consequences or costs of the risk. The likelihood of an accident has two components: the likelihood of the hazard occurring (with consideration for the hazard duration or exposure) and the likelihood of the hazard leading to an accident. Where these probabilities are quantified and independent, they can be multiplied to yield a single probability. These probabilities are rarely independent.

In risk analysis, there is uncertainty at all levels. Uncertainty arises because a model is always an incomplete representation of a system, and the parameters used to define the model contain measurement and epistemic uncertainty. Stochastic uncertainty in the system being modelled can be represented by means of probability distributions. But these distributions are again only best guess approximations of the underlying stochastic process. Uncertainties are sometimes difficult to quantify. As far as possible, the strength of knowledge of the models (including assumptions, simplifications, data and phenomena understanding) are to be assessed.

Complexity is an important source of uncertainty and hence risk. Complexity makes it difficult to understand or predict a system's behaviour, thus giving rise to epistemic uncertainty. The more complex a system is, the more opportunity there is for components to interact in unforeseen and possibly undesirable ways.

7.1.3. Performance-based Approach

A regulation can be either prescriptive (deterministic) or performance-based. A prescriptive requirement specifies features, actions, or elements to be included in the design or process, as the means for achieving a desired objective. Prescriptive requirements often do not consider risk or likelihood. They are frequently based on scenarios.

A performance-based approach focuses on a desired, measurable outcome. A performance-based approach focuses on identifying performance measures that ensure an adequate safety margin and offer incentives to improve safety and demonstrate that the intend of applicable requirements is met. A performance-based requirement relies upon measurable (or calculable) outcomes (i.e., performance results) to be met, but provides more flexibility to the licensee as to the means of meeting those outcomes.

7.1.4. A risk-informed, performance-based approach

The regulation based on an RIPB approach is an approach in which risk insights, engineering analysis and judgment (including the principle of defence in depth and the incorporation of safety margins), and performance history are used to (1) focus attention on the most important activities, (2) establish objective criteria for evaluating performance, (3) develop measurable or calculable parameters for monitoring system and licensee performance, (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes, and (5) focus on the results as the primary basis for decision making.

7.1.5. Risk Management Approach

The RIPB defence in depth approach provides the basis for the risk management (RM) approach. An RM approach is the process for identifying, analysing, and communicating risk and accepting, avoiding, transferring, or controlling it to an acceptable level considering associated costs and benefits of any actions taken.

The relation between safety objectives and RIPB defence in depth protection is to:

- Ensure appropriate barriers, controls, and resources to prevent, contain and mitigate exposure to radioactive material accounting to the hazard present, the relevant scenarios and the associated uncertainties;
- Ensure that the risks resulting from the failure of some or all of the established barriers and controls, including human errors are maintained acceptably low.

A key concept in the development of a risk management process is that technical analyses can be done in many ways, including traditional mechanistic analyses (e.g. thermal-hydraulic calculations), PSAs, and other techniques selected to support specific decisions related to particular issue and hazards.

The risk management approach needs to be consistent with principles of integrated risk-informed decision making provided in Ref. [21] and described in para 4.1.

7.2. BENEFITS OF APPLYING A GRADED APPROACH

As noted in Section 2.1, advanced nuclear reactors and installations vary tremendously, including the characteristics of the designs that provide protection against hazards. Existing LWRs, generally protected using active safety features, have applied the traditional safety and licensing approaches discussed in 7.1.1. However, as noted in this section, excessive conservatism or the imposition of requirements that do not result in a proportional benefit to safety or only add minimally to safety beyond an already existing adequate level of safety can be contrary to an efficient and effective regulatory framework. Overall, it is difficult to develop a single approach that can be applied to a full range of nuclear installations, including installations protected by active safety features, passive features or inherent design features, without applying excessive conservatism. This conservatism results in excessive time and cost for the installation design and site specific evaluation.

Application of a graded regulatory framework/approach, supported by RIPB strategy, is essential to fully understand the real margins to safety that exist in nuclear installations. As noted in Sections 5 and 6, the approach focuses the analysis and nuclear installation protection resources on risk-significant contributions. Application of a deterministic/traditional approach may divert resources from important safety issues while increasing costs to licensees and thus to consumers, without a matching safety or security benefit. As noted in Section 5, during the design phase, design decisions could develop a robust design for SSCs related to inherent passive safety functions in a way that their contribution to the nuclear installation risk related to external and internal hazards remain very low (not significant). When applied to the licensing framework, a similar focus can be used to focus the regulatory requirements and required analysis based on the contribution to the nuclear installation risk related to external and internal hazards. As a result, different nuclear installation designs and even different sites would have different sets of resulting requirements, in large part based on the results from the site specific hazards analysis when applied to a specific design.

Finally, the benefits of application of the methodology presented here are expected to be larger for advanced reactors prior to construction, with designs incorporating advanced safety features. A graded regulatory approach would therefore encourage innovative design approaches, which would in the long run simplify the licensing approach, while lowering both the risk and the cost of the nuclear installation construction and operation. As discussed in Section 6.1, when applying this approach for selection of a reactor design for a given site, the selection process can be influenced by the nuclear installation which is best designed for a given set of site specific hazards.

8. CONSIDERATIONS FOR FUTURE WORK

The objectives of this publication are mainly intended to provide a framework for optimization of safety measures for protection of advanced reactors (such as SMRs) utilizing a RIPB process and application of a graded approach for:

(a) Development of the technology-neutral safety framework for assessing the applicability of site evaluation requirements considering site–installation interactions, and;
(b) A methodology for an overall optimization of safety measures for protection against external hazards supporting effective and balanced implementation of the defence in depth concept.

The publication is supported by annexes providing examples of real cases where a RIPB approach was used to optimize the safety measures for protection in relation to the design for external hazards and showing how a graded approach can be used during the site evaluation phase for hazards characterization. This framework is also discussed in the context of various regulatory approaches.

The main elements of this framework are not new and have been used for various applications in the last two decades. Most of advanced reactor designs use DSA and PSA during the design development for confirming that the intent of safety requirements and the adequacy of the defence in depth were achieved.

The proposed framework considers site installation interactions and proposes the application of a graded approach for site evaluation in a manner that will maintain confidence in the adequacy of site specific hazards parameters resulting in adequate safety margins and risk performance targets in relation to site specific external hazards.

Many advanced reactor designs incorporate passive safety functions that contribute to enhancement of the overall safety of the installation. Many passive safety systems are related to structural integrity and structural performance in general and against external hazards in particular, since external hazards could challenge their structural integrity and structural performance. Therefore, the protection of these systems and structures against external hazards became more important due to the larger contribution of external hazards to the overall nuclear installation risk. Also, the limited or no redundancy of some passive safety systems may necessitate larger safety margins for reducing potential risk contribution. From this perspective using a risk informed approach for assessing risk contribution against various defence in depth challenges helps to optimize the safety measures for protection in relation to the design and to define adequate performance targets related to these passive safety systems.

The advanced designs are basically developed for bounding various site conditions (not just a specific site). This may provide a source of additional safety margins if the bounding design parameters are significantly more severe than that of specific site parameters. This additional safety margins can be carefully used in a graded approach process for site evaluation verifying that sufficient safety margins remains available, and the risk performance targets are not compromised.

Future work is aimed to provide more technical guidance for implementation of such a framework as more international experience continues to accumulate in relation to design license and construction of advanced reactors.

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ANNEX I.
EXAMPLE OF RISK-INFORMED APPROACH: PRISM REACTOR DESIGN OPTIMIZATION OF SAFETY MEASURES AGAINST INTERNAL AND EXTERNAL HAZARDS

I–1. INTRODUCTION

This annex describes the use of the PRISM PSA[1] to develop an approach for potential optimization of safety measures for the plant design against external hazards. This process utilizes the previously developed PRISM PSA which included a comprehensive analysis for internal events and a scoping evaluation for external hazards that screened-in and screened-out potential internal and external hazards. This annex includes a description of the PRISM design, including use of passive reliability features that result in an overall low risk to the public in comparison to existing LWRs.

The PRISM example presented in this annex is an example of an advanced NLWR reactor, with an overall low risk profile and a resulting small safety footprint. The use of passive safety systems results in the plant being well positioned to respond to any event that impacts off-site power or leads to a station blackout – which may occur as a result of a severe external hazard. The small safety footprint means that there is a much smaller number of potentially safety related components, which translates into a small number of components that need to be protected or robust against unscreened external hazards.

This is demonstrated in two example cases of hazard optimization of safety measures, which result in a small number of components that are required to have enhanced capability against the unscreened hazards. As a result, optimization of safety measures is less complicated than would be performed for an existing LWR. In the case of seismic response, enhanced capability for passive cooling using a Reactor Vessel Auxiliary Cooling System (RVACS) and for the vessel and internals would be the focus of optimization of safety measures, given that a reliable reactor trip can be ensured.

I–1.1. PRISM Plant Overview

The PRISM reactor is a pool-type, metal-fueled, small modular Sodium Fast Reactor (SFR). PRISM employs passive safety systems, digital instrumentation and control, and modular fabrication techniques to expedite plant construction. The PRISM has a rated thermal power of 840 MW/h and an electrical output of 311 MW/h. Each PRISM module has an intermediate sodium loop that exchanges heat between the primary sodium coolant from the core with water/steam in a sodium–water steam generator. The steam from the sodium–water steam generator feeds a conventional steam turbine. A diagram of the PRISM nuclear steam supply system is shown in Fig. I–1.

[1] This Annex uses the term PSA and PRA interchangeably.

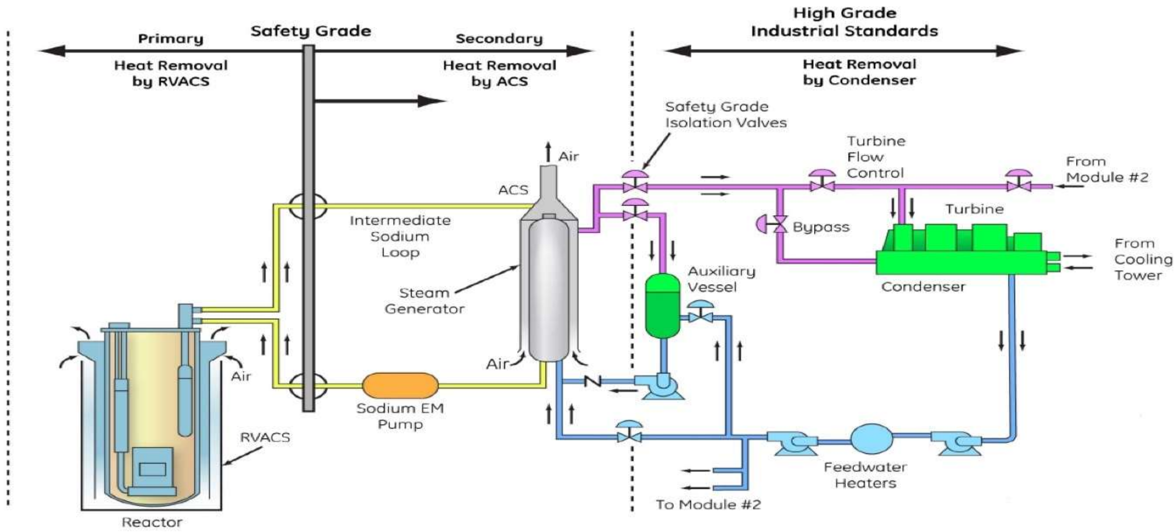

FIG. I–1. PRISM nuclear steam supply system.

Two reactor units are paired to form one power block. A PRISM power block is shown in Fig. I–2. The power block supplies steam for one 622-MW turbine-generator. The commercial PRISM plant achieves a high-capacity factor by utilizing six reactor modules and their associated steam generating systems arranged in three identical power blocks. An unplanned outage in one PRISM module or power block does not impact the plant electrical output as dramatically as it does in a large single-unit site.

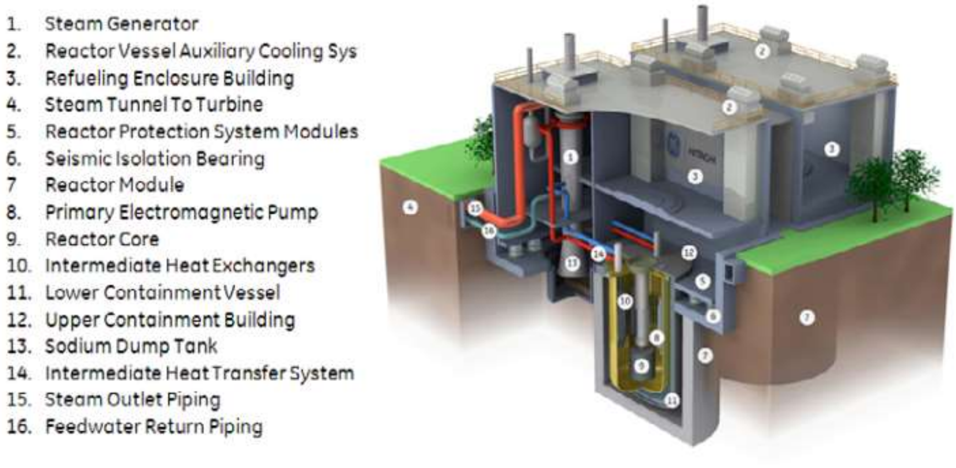

FIG. I–2. PRISM power block.

Plant electrical output can be tailored to utility needs by the modular addition of power blocks. This modularity allows expansion from one power block to as many as desired by the utility on one site.

I–1.2. PRISM PSA Project Overview

The original PRISM PSA was developed in the 1980s and is documented in Refs [I–1, I–2]. The PRISM project began in 1984, and the preliminary safety information document (PSID) was first submitted in 1986 (with several revisions following). The US Nuclear Regulatory Commission (NRC) reviewed the PSID, including the PSA, and responded with NUREG-1368 in 1989 [I-2], with continued revisions until 1994. However, US Department of Energy funding for the programme was terminated in the early 1990s. The original PRISM PSA includes a limited seismic PSA evaluation. The PRISM PSA was one of several PSAs performed on U.S. SFRs, as discussed in Ref. [I–3]

GE Hitachi Nuclear Energy (GEH) teamed with Argonne National Laboratory to perform research and development of next-generation Probabilistic Risk Assessment (PRA) methodologies for the modernization of an advanced non-LWR PRA. This project is supported by the Department of Energy under Award Number DE-NE0008325. This effort builds upon a PSA developed in the early 1990s for GEH's PRISM SFR discussed above.

Fig. I–3 provides an overview of the project's deliverables and benefits. This next-generation PRA allows a risk-informed research and development prioritization option going forward. GEH, with its experience in PRA, led the research efforts and leveraged Argonne's expertise in advanced PRA analysis.

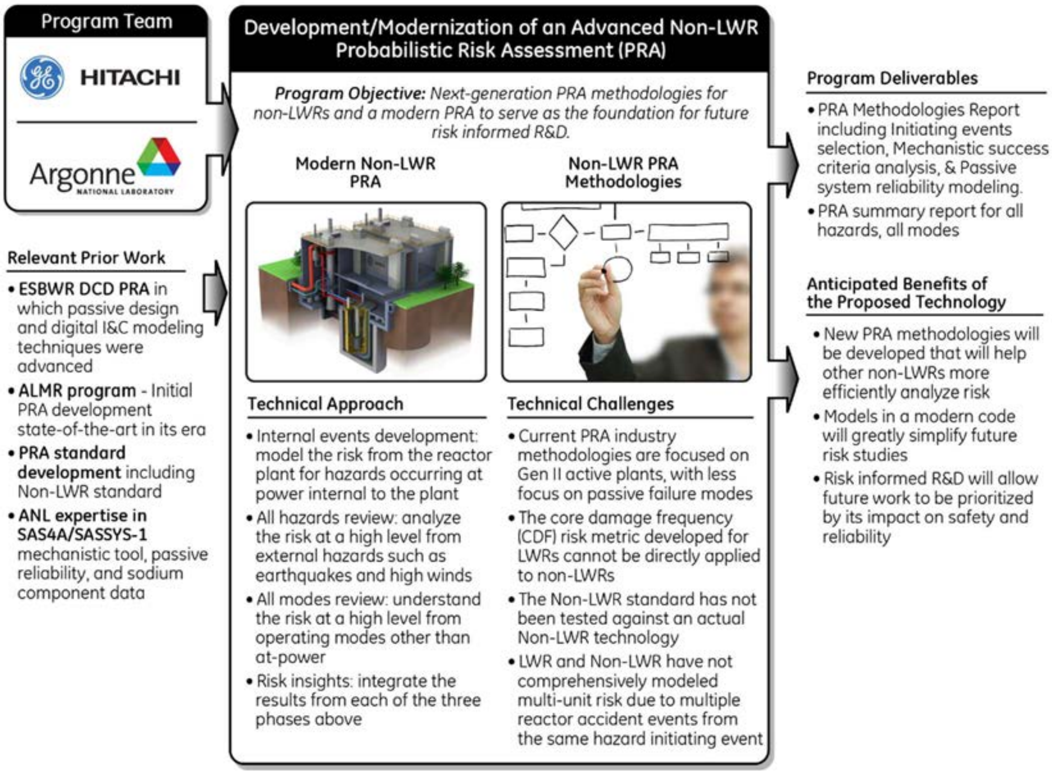

FIG. I–3. PRISM PRA modernization project overview.

I–1.3. PSA Methodology Overview

The PRISM PSA update was performed as described in Section I–1.2 above in an iterative manner, yet there was a linear flow of information from one element to the next. The relationships between PSA tasks are summarized in Ref. 1 and there is a general delineation between tasks associated with modelling the frequencies of accident conditions and those evaluating the consequences of the accidents. A visual representation of these relationships is provided in Fig. I–4.

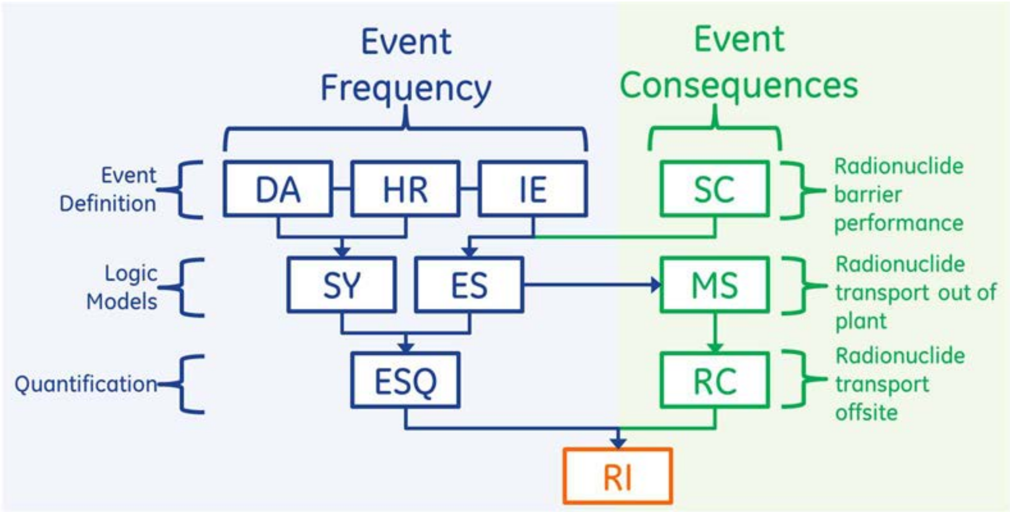

FIG. I–4. PRISM PSA methodology.

Many of these elements were developed using methodologies typical to LWR PSA, however several new or modified approaches were utilized especially in the Success Criteria (SC) and Mechanistic Source Term (MS) elements. The Initiating Event (IE), identification relied upon LWR experience, Non-LWR experience and an inductive systems-based approach. In the Data Analysis task, the data pedigree was established using a prioritized data source selection process in which design specific data was compared against more plentiful generic nuclear and non-nuclear sources. Finally, concerning the Human Reliability Analysis task, although non-LWRs are less reliant on operator actions, pre and post initiator human reliability is still an important PSA element and can identify potential design weaknesses. Human failure events were identified and error probabilities applied accordingly.

For the Event Sequence analysis, three general groups of event trees were developed to analyse event progression from IE to release:

- Protected: sequences in which active reactivity insertion is successful via the control rods;
- Unprotected: sequences where control rod insertion fails and Inherent Reactivity Feedbacks or the ultimate shutdown system must succeed to satisfy the reactivity control safety function;
- Confinement: analyse the various radionuclide barrier success and failure combinations for both protected and unprotected sequences.

The identified active and passive mitigating system functions and features from the Event Sequence element were then modelled in the Systems Analysis task. For active systems, traditional fault tree analysis was performed, while passive systems and features were modelled using state-of-the-art passive system reliability modelling techniques.

The overall Success Criteria for the PRISM design defined the combinations of barrier and mitigating systems successes needed to prevent release, based on the identification of possible release categories. Each release category consists of a quantified level of fuel barrier damage and a combination of intact and failed confinement release barriers. For each event sequence modelled, plant parameters are defined with various thresholds that represent the different release categories that are possible for that sequence. Based on this analysis, the PSA safety functions were defined for PRISM as shown in Fig. I–5.

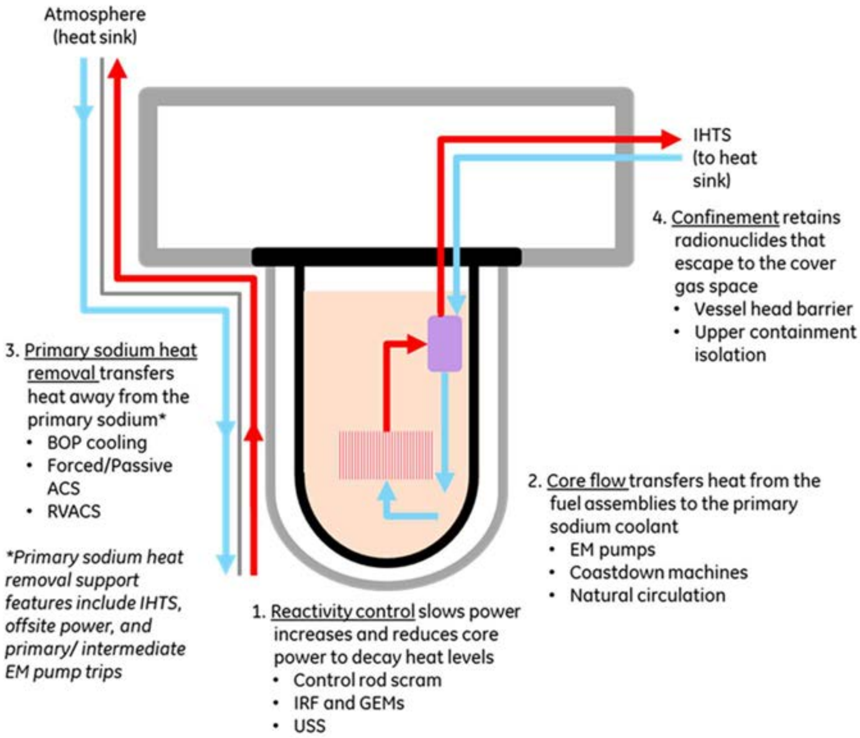

FIG. I–5. PRISM PSA safety functions[2].

The Mechanistic Source Term (MS) and Radiological Consequence tasks analysed radionuclide progression for the various event sequences from the fuel to offsite through the following radionuclide barriers:

(1) Fuel: The metal fuel retains many radionuclides (i.e., plutonium, neptunium) within its matrix as long as the fuel has not melted;
(2) Cladding: The fuel cladding provides a barrier for gaseous fission products (i.e., xenon, krypton) as long as the cladding is intact;
(3) Sodium: The sodium coolant acts as a third radionuclide barrier by retaining fission products either by plate-out, chemical solubility or adsorption mechanisms;
(4) Vessel: The reactor vessel is the radionuclide barrier for fission gases that are released by the sodium to the cover gas space with cladding failure;
(5) Containment: The containment is the final radionuclide barrier and retains fission products with vessel leak-by failure (e.g. head seal leak).

The final step, Risk Integration, combines the results of the Event Sequence Quantification (ESQ) and the Radiological Consequence analysis. For each release category identified in the PSA, the ESQ gives an annual probability of occurrence, and the Radiological Consequence gives the consequence of occurrence. Combining these two elements gives quantitative

[2] Safety Functions include use of both Safety Related and non-safety related systems for performing the function. Definitions for acronyms used in this figure: BOP: Balance of Plant, ACS: Alternate Cooling System, RVACS: Reactor Vessel Alternate Cooling System, IHTS: Intermediate Heat Transport System, EM Pump: Electromagnetic Pump, IRF: Inherent Reactivity Feedback, GEMs: Gas Expansion Modules, USS: Ultimate Shutdown System

measures of risk that can then be compared to health goals for site workers and the public. These results are presented in the section that follows.

A subset of this effort is the development of PSA methodologies for the creation of the reliability database (RDB) [I–4]. The goal of the RDB methodology is to provide a systematic procedure to review available reliability data for its applicability to the PRISM PSA, and derive reliability estimates for the quantification of initiating event frequencies and system reliability. This publication provides details on the developed RDB methodology and how the methodology satisfies the requirements of the ASME/ANS PRA standard for advanced non-LWRs.

I–1.4. Hazard Scoping Analysis

The PRISM PSA Hazard Identification and Analysis process is used to perform an efficient, yet holistic vulnerability study across all hazards and all functions credited in the full-scope PSA. It provides process steps and an application example that uses as input the comprehensive list of hazards developed in the hazard identification task. Combining the hazards list with the list of full-scope PSA functions then allows potential vulnerabilities to be assessed. With potential hazards numbering in the hundreds and a number of PSA functions, there will likely be thousands of combinations of hazard–function pairs. Despite the high number of hazard–function pairs, the process performed for the PRISM PSA can be performed efficiently since vulnerability patterns will emerge due to similarities across certain hazards. After defining screening criteria, hazards can be screened out based on qualitative or quantitative criteria.

Table I–1 [I–1] provides the process steps for this task with a more detailed description.

TABLE I–1: HAZARD IDENTIFICATION, POTENTIAL IMPACT, AND QUALITATIVE SCREENING PROCEDURE

Step	Description
1	Identify hazards
2	Assess SSCs/function vulnerability for each hazard
2a	Develop full-scope safety functions list
2b	Create hazard–safety function matrix
2c	Identify potential vulnerabilities
3	Identify screening criteria (qualitative and quantitative)
4	Perform qualitative screening
5	Document results and basis

For the qualitative screening portion of the scoping hazard analysis methodologies, the step is applicable for a PRISM reactor with an air-cooled passive heat removal system. The passive features for PRISM reveal that many hazards typically considered not applicable for LWRs, may be more relevant for non-LWRs (e.g. outside environment extremes). Graphically, the process follows the procedure outlined in Fig. I–6 [I–1]. The remainder of this section describes each step of the process and includes a simplified example.

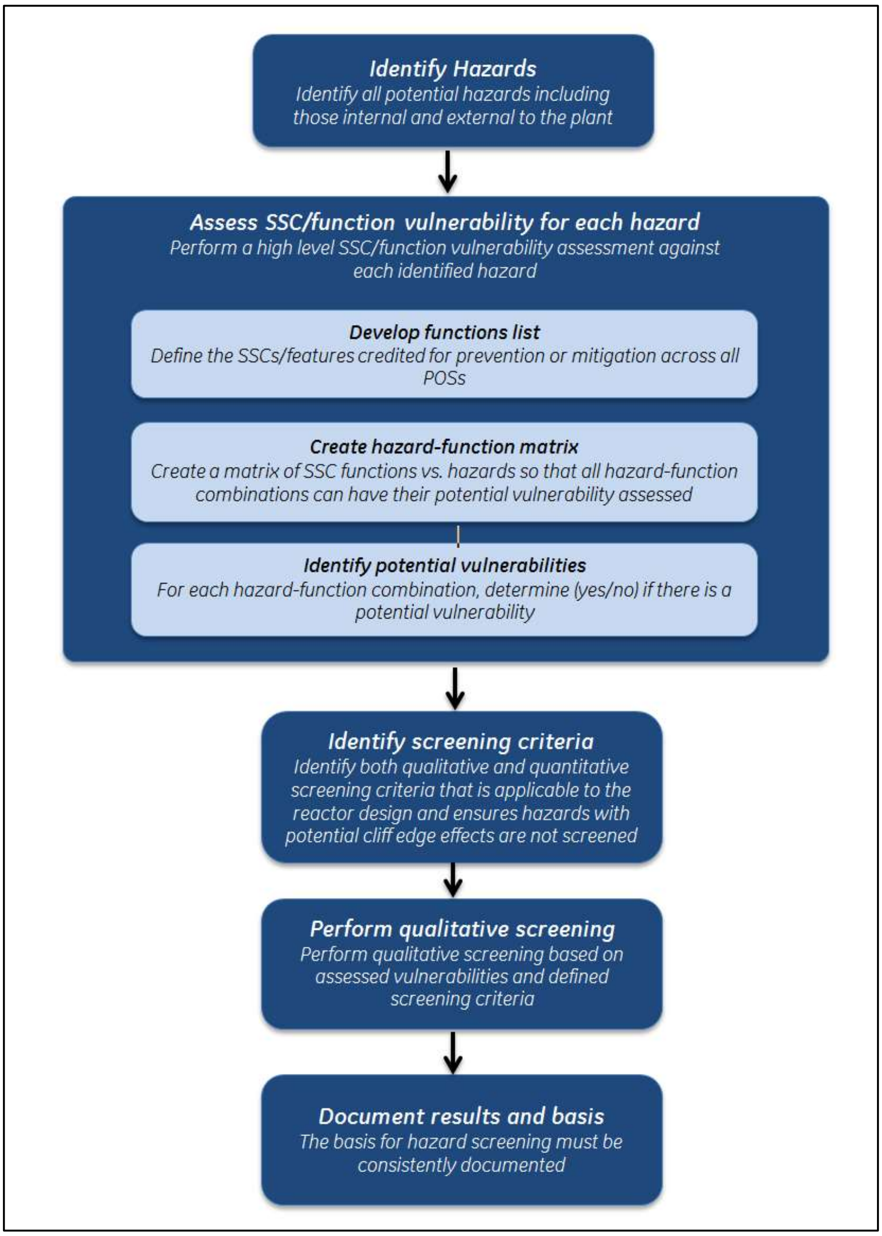

FIG. I–6. Qualitative screening procedure.

The first step of the process is to identify all potential hazards. As discussed in Ref. [I–5], numerous sources are available to ensure a comprehensive list of hazards is developed. For LWRs, well established hazards such as internal fire, internal flood, and seismic are not eligible for screening and therefore not typically included in this list. For a scoping study for non-LWRs, however, inclusion of these hazards is recommended even though they will be retained through the screening process. Their inclusion allows the scoping study to show all potential hazards and the high level plant vulnerability to them.

The list will likely exceed dozens of hazards, but to simplify this example, only a partial list of hazards will be presented. Table I–2 [I–1] contains several example hazards found in various hazard identification sources including Appendix 4.5.11-A in Ref. [I–6].

TABLE I–2. EXAMPLE LIST OF HAZARDS

Hazard
Aircraft impacts
External Flooding
High summer temperature
Extreme winds and tornadoes
Sandstorm
Extreme Snow
Snow (drift)
Forest fire
Seismic
Internal Fire
Internal Flood
Landslides
(...)

The lists from the Step 1 and Step 2a above can be combined to form the hazard–function matrix. The layout for the SFR example application is presented in Table I–3 [I–1].

TABLE I–3. HAZARD-FUNCTION MATRIX

SSC/features providing safety functions

Potential hazards

	CR Insert	Reactivity Feedback	Core Flow	Air Cool	SD Cool	Vessel	Contain.	I&C	Human	(...)
Aircraft impacts										
External Flooding										
High summer temperature										
Extreme winds and tornadoes										
Sandstorm										
Extreme Snow										
Snow (drift)										
Forest fire										
Seismic										
Internal Fire										
Internal Flood										
Landslides										
(...)										

With the matrix established, potential vulnerabilities (including potential cliff edge effects) can be identified. For the example SFR application, each hazard–function pair is examined and conservatively assessed for vulnerabilities. A simple code is entered in each cell along with an appropriate background colour so that vulnerabilities can be visualized horizontally (a single hazard across all functions) or vertically (a single function across all hazards). Cells coloured in green have no identified vulnerability, those in yellow are potentially vulnerable for a single unit, while those in red are considered multi-unit vulnerabilities. Table I–4 [I–1] shows the vulnerability assessment for four hazards.

TABLE I–4. COMPLETED HAZARD-FUNCTION MATRIX FOR FOUR HAZARDS

	CR Insert	Reactivity Feedback	Core Flow	Air Cool	SD Cool	Vessel	Contain.	I&C	Human	(...)
Internal Fire	**N3**	**N3**	**Y1**	**Y1**	**Y1**	**Y1**	**Y1**	**Y1**	**Y2**	
External Flooding	**N3**	**N3**	**Y4**	**N3**	**Y4**	**N3**	**N3**	**Y4**	**Y2**	
Sandstorm	**N3**	**N3**	**N3**	**Y8**	**N3**	**N3**	**N3**	**N3**	**Y2**	
High summer temperature	**N5**	**N5**	**N5**	**Y7**	**N3**	**N5**	**N5**	**N5**	**N6**	
(...)										

For each unique cell code entered in the matrix, an accompanying justification is to be recorded. Table I–5 [I–1] shows how each assessment in Table–4 was determined.

TABLE I–5. VULNERABILITY ASSESSMENT JUSTIFICATION

Cell Code	Justification
Y1	Fire could cause failure of function due to direct effects or through spurious operation.
Y2	Human reliability could be impacted by this hazard.
N3	These hazards would not affect these systems/functions since they are not exposed to the hazard or are robust enough to withstand the hazard.
Y4	External Flooding could potentially cause these systems/functions to fail if submerged or due to spray effects.
N5	High summer temperature is not expected to affect these functions since the plant's Heating, Ventilation and Air Conditioning (HVAC) system would maintain appropriate temperatures inside the plant.
N6	Human reliability is not expected to be impacted by this hazard.
Y7	Exceptionally high summer temperatures could potentially degrade the performance of passive air-cooling systems since they utilize outside air.
Y8	A sandstorm could degrade the passive air-cooling systems since they utilize outside air.
(...)	

Based on the above, and the application of additional qualitative screening criteria discussed in Ref. [I–6] (not covered in detail here), many of the identified hazards and hazard combinations can be screened from further analysis for the PSA.

Qualitative screening provides an opportunity to significantly reduce the number of hazards to evaluate quantitatively in subsequent steps. Many of the hazards typically eliminated in an LWR PSA, however, may need to be retained for a non-LWR PSA.

For this example, the presence of the SFR’s passive air cooling system requires typically screened-out hazards such as sandstorm, snow, and forest fire to be retained due to a potentially degraded heat removal performance from these hazards. For example, Nonmandatory Appendix 4.5.11-a of Ref. [I–6] lists EXT-B1 screening criteria 1 & 4 as applicable to the sandstorm hazard and remarks that it is “Included under extreme winds and tornadoes potential blockage of air intakes with particulate matter is generally considered in plant design.”

This screening criteria is reasonable for LWRs that don’t use passive air cooling as the ultimate heat sink, but for PRISM, a sandstorm could reduce the heat transfer in the passive air cooling system that uses outside air as its cooling source (e.g. full blockage is not needed to impact the safety functions). If severe enough this or other similar hazards may increase the plant’s overall risk (future research would be required to determine) since the air cooling system is a key heat removal system.

Several hazards from the example hazard list above can still be screened out based on the qualitative criteria. Retained hazards (unscreened) are then analyzed either in a bounding fashion or using detailed analysis. The bounding assessment includes determining a hazard frequency (or frequency/magnitude curve) and performing a bounding risk assessment using demonstratively conservative assumptions. Once complete, additional screening can be performed if the bounding analysis results in a hazard risk below established quantitative screening criteria. A simplified procedure for this process is shown in Fig. I–7 [I–1].

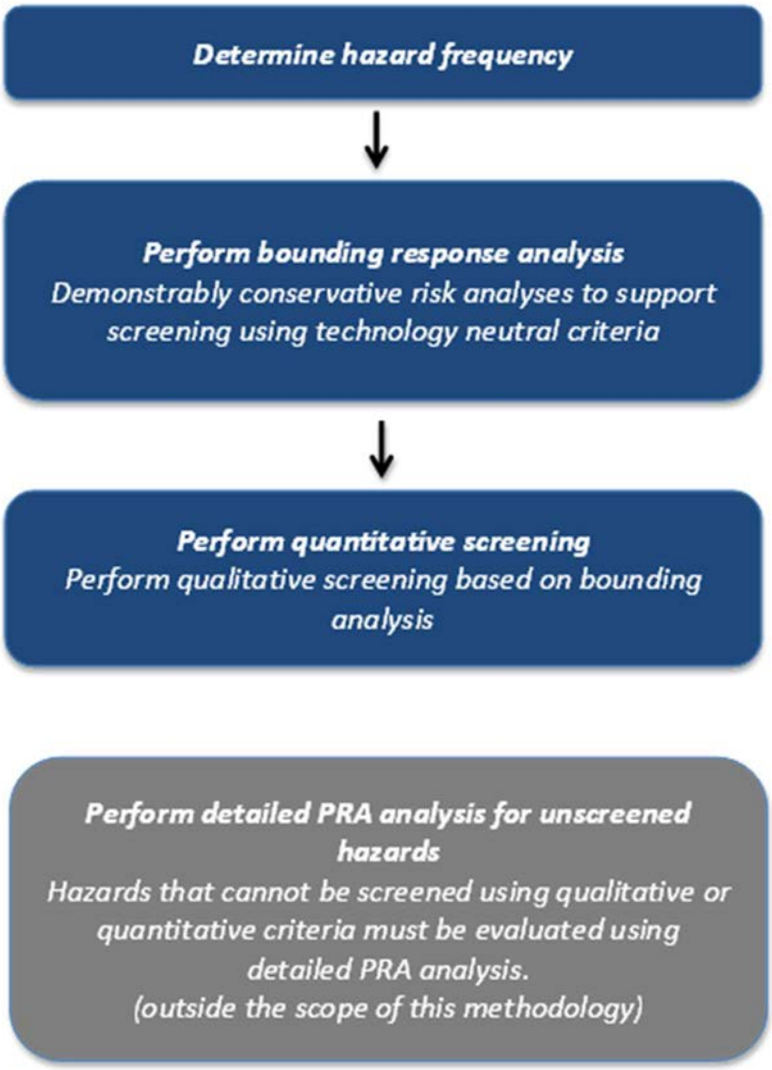

FIG. I–7. Quantitative screening procedure.

I–2. PRISM PROBABLISTIC SAEFTY ASSESSMENT RESULTS SUMMARY

Summary of PRISM PSA results is provided in following subsections.

I–2.1. Internal Events PSA Results

Risk metrics developed for the PRISM PSA revolve around the release categories defined in the Mechanistic Source Term element. The overall plant risk can be described using a number of metrics, where each metric is essentially a frequency-weighted average of the consequences of postulated events in a given release category. Traditional risk metrics, such as CDF, are not meaningful for PRISM or other non-LWR designs (as recognized by the trial use PSA standard) since they assume LWR core, reactor, and containment design features. Therefore, risk for PRISM has to be expressed directly in terms of offsite consequence measures as described below.

Reference [I-6] uses technology-neutral risk metric terms, and provides the following discussion and background for this approach.

The current LWR PSA standards are based on risk metrics such as CDF and LERF that have been defined in terms of LWR-specific characteristics. CDF and LERF are also known as "surrogate risk metrics" for a more complete set of risk metrics that are produced in a Level 3 PSA. For advanced non-LWRs, which include diverse reactor types such as HTGRs, LMRs, and other advanced Generation IV reactor concepts, a technology-neutral approach has been adopted in consideration of the fact that a technology-neutral definition of core damage does not exist. The risk metrics used with this technology-neutral approach are the standard risk metrics used in LWR Level 2 and Level 3 PSAs.

Three major off-site consequence-related goals are established based on Ref. [I-7]. These QHOs are defined in the sections that follow along with a presentation of the PRISM risk results. Note that these results are for one PRISM power block – two reactor modules connected to one balance of plant – and consider single and common cause initiating events whose event sequences occur asymmetrically[3]. Results do not include external hazards or potential hazards during other plant operating states.

I–2.2. Individual Risk QHO and PRISM Results

QHO: The risk of prompt fatalities that might result from reactor accidents to an average individual in the vicinity of a nuclear power plant should not exceed one tenth of one percent (0.1%) of the sum of "prompt fatality risks" resulting from other accidents to which members of the U.S. population are generally exposed [I–7].

As noted in Ref. [I–7], "vicinity" is defined as the area within 1.61 km (1 mile) of the plant site boundary. "Prompt Fatality Risks" are defined as those risks to which the average individual residing in the vicinity of the plant is exposed to as a result of normal daily activities. "Other accidents" are used to represent the sum of risks that result in fatalities from such activities as driving, household chores, occupational activities, etc.

For this evaluation, the sum of prompt fatality risks is taken as the U.S. accidental death risk value of 39.1 deaths per 100,000 people per year [I–8]. The PRISM internal events at-power individual risk is represented in the frequency-consequence curve shown in Fig. I–8 [I–1]. This curve shows the QHO goal and the prominent PRISM internal event release category consequence results (the blue data points in the figure).

[3] Concurrent event sequences are evaluated separately in the multi-unit analysis.

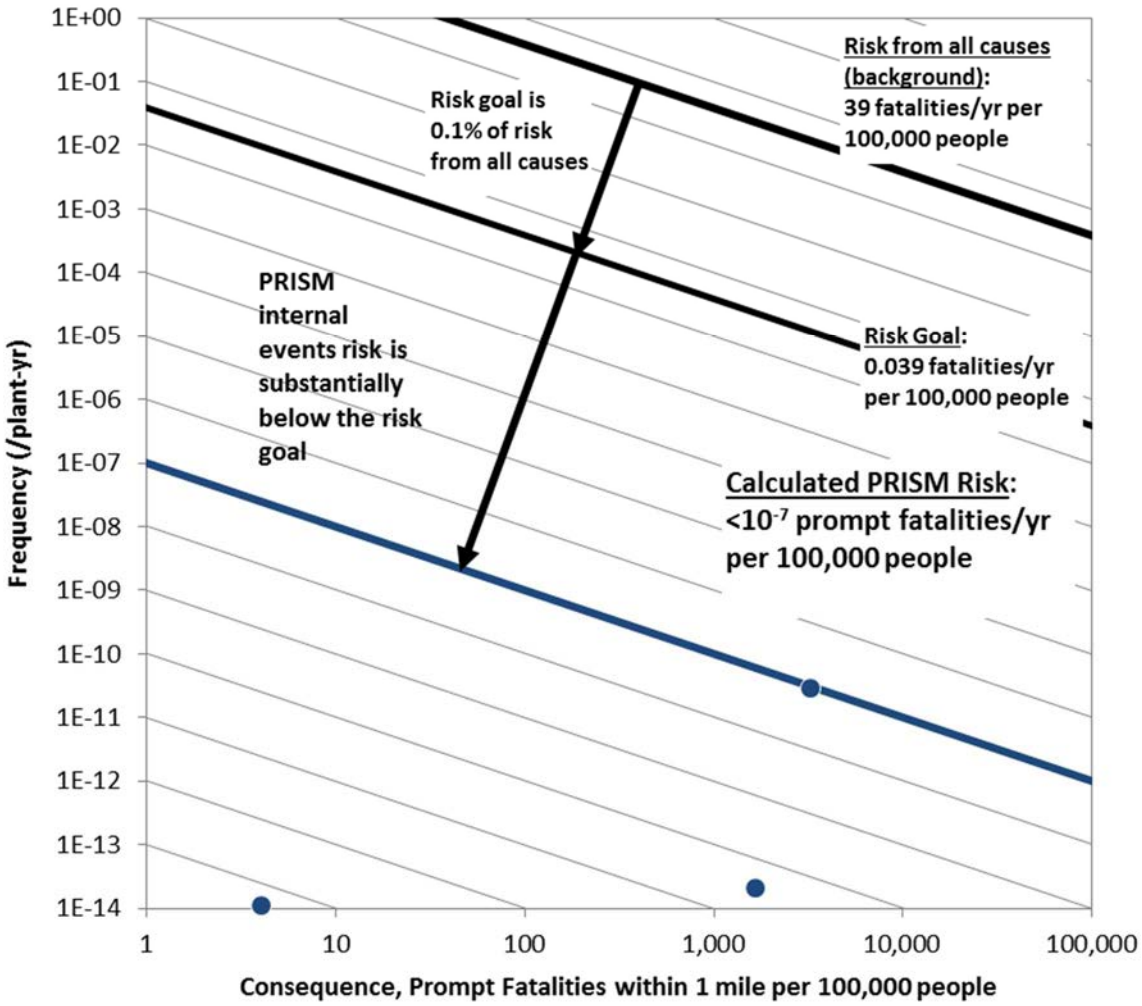

FIG. I–8. Comparison of individual risk to risk goal.

I–2.3. Societal Risk QHO and PRISM Results

QHO: The risk of cancer fatalities that might result from nuclear power plant operation to the population in the area "near" a nuclear power plant should not exceed one tenth of one percent (0.1%) of the sum of the "cancer fatality risks" resulting from all other causes to which members of the U.S. population are generally exposed. [I-7]

As noted in Ref. [I-7], "near" is defined as within 16.1 km (10 miles) of the plant. The "cancer fatality risk" is taken as 169 deaths per 100,000 people per year based upon 1986 statistics [I–9]. The PRISM internal events at-power societal risk is represented in the frequency-consequence curve in Fig. I–9 [I–1]. This curve shows the QHO goal and the prominent PRISM internal event release category consequence results (the blue data points in Fig. I–9).

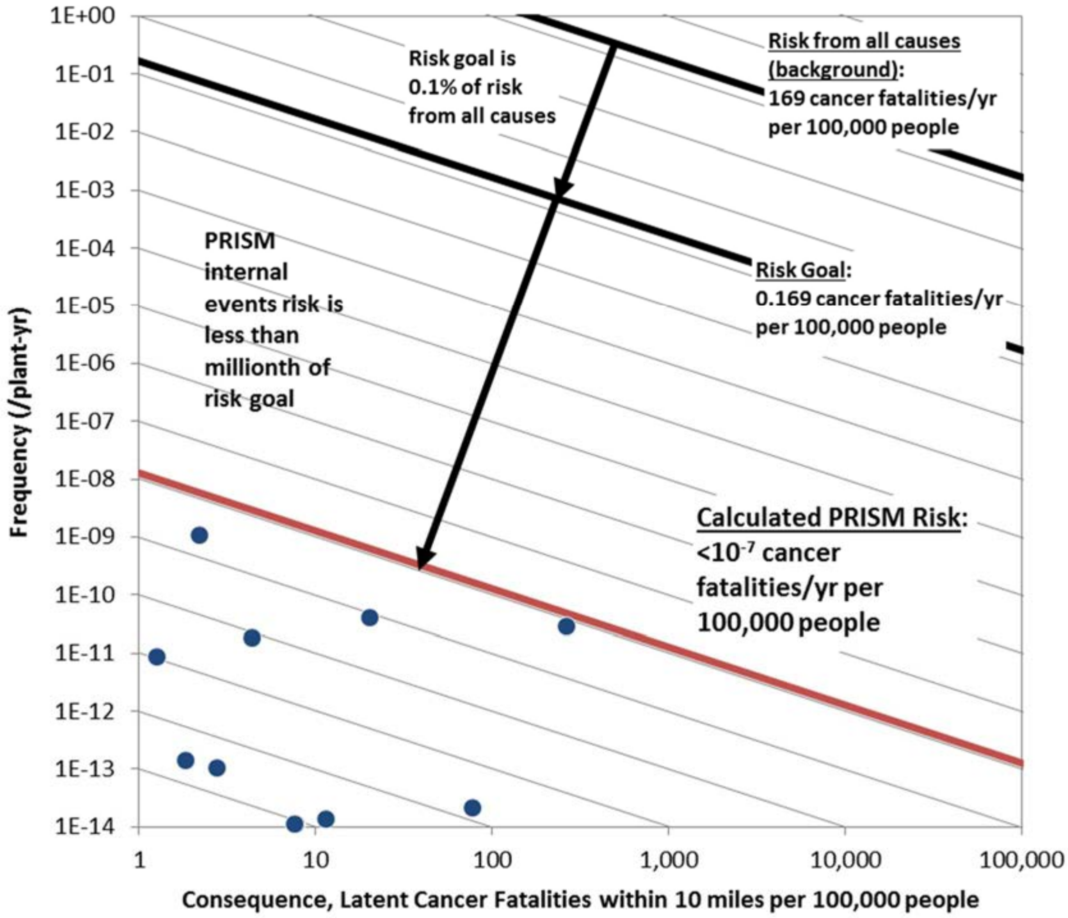

FIG. I–9. Comparison of societal risk to risk goal.

The PRISM PSA results were utilized for a pilot application of the Licensing Modernization Project (LMP) [I–10]. The LMP results included the selection of licensing basis events (LBEs) and safety related SSCs/Functions. The LBEs selected were based on the PRISM PSA internal events analysis. The resulting list of safety related SSCs included the following:

- Digital I&C such as the reactor protection system (RPS);
- Control rods and drives;
- Primary electro magnetic (EM) pump breakers (EM pumps are required to trip on a reactor trip);
- Reactor vessel internals and vessel;
- Reactor vessel auxiliary cooling system (RVACS);
- Supporting low-voltage power to the above (DC Power and 120VAC).

The most noticeable result from the scope of the safety related selection analysis is the small number of safety related SSC functions. The list does not include the typical LWR list of safety related SSCs such as the reactor coolant piping (the reactor coolant does not leave the vessel), emergency power, injection pumps, and active cooling such as steam generator (SG) cooling for a pressurized water reactor (PWR). When reviewing the related SSCs list from the viewpoint of hazard vulnerability, the results can be viewed from two initial plant responses:

(1) Short term plant response:

(a) Plant trip needs to occur resulting in rod insertion (via gravity drop);
(b) Primary EM pumps need to trip.

(2) Long-term heat removal function:
(a) RVACS needs to remove heat through passive cooling (air cooling);
(b) The reactor vessel and internals need to remain functional.

Power is generally only needed to support EM pump breaker opening, and to support the operator monitoring of the reactor status.

I–2.4. Hazard Scoping Evaluation Results

The PRISM PSA did not fully quantify all hazards as was performed for the internal events PSA. The quantification process was performed sufficiently to determine which would need to be evaluated for a detailed quantification, when applied to either a generic site or for site specific analysis. The retained hazards for the PRISM PSA included:

- Seismic events;
- Degraded heat sink (RVACS) events such as extreme weather, ice storms, smoke events, etc.;
- Internal flooding;
- Internal fires;
- Heavy Load Drops;
- Catastrophic external impacts such as aircraft impact;
- Hazards affecting automatic or manual actuation, such as chemical or hazardous gas releases.

Heavy load drops are generally covered under the all-modes PSA, as a separate initiating event that can occur during outages. This potential initiating event is not discussed further here. The remaining hazards generally include hazards that can potentially impact the main short term or long term safety related functions discussed above, with the short term response events involving the last bullet above. These hazards are expected to be lower risk overall and prevented by site selection and location (generally minimized for both existing and advanced reactor site selection). As a result, the main focus for hazard impact involves the response of long term safety related functions including the vessel and internals, and RVACS.

I–3. USE OF THE PROBABLISTIC SAFETY ASSESSMENT RESULTS TO OPTIMIZE SAFETY MEASURES OF THE PRISM DESIGN

I–3.1. Example Optimization of Safety Measures for Screened Hazards

The hazards listed in Section I-1.4 do not include several hazards typically included in LWR hazard PSAs. Two examples include:

- External flooding;
- Hazards impacting circulating water or service water.

Both of these result in loss of the normal heat removal through the SGs and the condenser. External flooding can also affect other systems supporting plant functions, such as electrical power, if the flooding exceeds the protection for these systems. However, given a plant trip (control rods insertion) will reliably occur following both an external flooding event and events affecting circulation or service water, then the protection against these events is provided by (a)

RVACS removing decay heat and (b) the vessel/internals being un-affected by the hazards. The loss of circulating or cooling water does not impact either safety function. Additionally, since RVACs intakes and outlets are on the building roof, generally it is not expected that external flooding would impact RVACS. However, for a site specific installation of PRISM, this assumption would need to be confirmed to ensure external flooding hazards remain screened. The optimization of safety measures for protection against external flooding would potentially include the need to ensure RVACS operation is not impacted by external flooding events that may impact the site.

I–3.2. Example Optimization of Safety Measures for Unscreened Hazards

The optimization of safety measures for protection against unscreened hazards would generally involve a similar evaluation. This includes to ensure both the short term and long term functions are either not impacted by the hazard or sufficiently reliable to ensure the risk to the plant is low. Likely, the most challenging hazard for any advanced reactor and for PRISM would be seismic events. The optimization of safety measures for protection against seismic hazards would include:

(1) Reactor Trip – ensure a reactor trip occurs with high reliability either before the seismic hazard fully impacts the site (e.g. using a seismic trip system) or during the event. If the later is implemented in the design, the control rod insertion function would require a high seismic capacity, as supported by a highly reliable RPS that would function to remove power to the control rods and utilize the gravity drop function.
(2) RVACS operation and reactor vessel/internals – the overall low risk of the plant can be ensured only if the RVACS and vessel have a high seismic capacity, ensuring both remain functional following a range of seismic events. Both are easily designed to have an overall high seismic capacity. However, if the overall risk for PRISM is to be much lower than existing reactors, as indicated by the risk calculated by the PSA QHOs discussed in Section I–2.1, the capacities of these functions would need to be higher that for existing reactors. The capacities required would depend on the site selected.

As discussed, optimization of safety measures for protection against seismic hazards involves a number of factors which need to be considered to ensure the risk from seismic events is sufficiently low including consideration of uncertainty. The PRISM design, which provides an overall small list of safety related SSCs and functions, would be relatively simple to optimize safety measures for protection. Focusing on the RVACS and vessel/internals, the following options for optimization of safety measures would be considered:

(1) The site specific installation would require the development of a site specific hazard characterization, as performed using the site SSHAC evaluation. Most sites would perform a SSHAC level 3 evaluation. However, if the analysed seismic risk for the plant was low (e.g. well below the QHOs or acceptance criteria for the site), a lower SSHAC level evaluation (e.g. SSHAC level 2) may be proposed (See also Annex II). This may save a lot of money for the analysis, and up to a year or more of schedule time.
(2) The implementation of the above lower analysis cost and schedule would require, at a minimum, the design implementation of a robust RVACS and vessel/internal design. The specific capacity of each would depend on the site, but well above the robustness estimated from existing sites or using generic fragility analysis. The cost

of these design improvements would need to be estimated and compared with the potential cost savings for the reduced analysis costs. However, when considering the tradeoff, the increased plant robustness for safety related functions would result in direct risk reduction for the design, while the increased plant analysis for the seismic SSHAC evaluation would result in reduced uncertainty in the site hazard curve. Generally, a reduced plant risk would be more valuable, given an acceptable level of uncertainty in the hazard curve.

In view of above optimization of safety measures, the implementation in the design requirements and construction costs would need to be considered. The increased capacity of RVACs and the vessel/internals would result in higher construction costs, as well as increased requirements for each.

I–3.3. CONCLUSIONS

PRISM is an example of an advanced non-LWR reactor, with an overall low risk profile and a resulting small safety footprint. The use of passive safety systems results in the plant being well positioned to respond to any event that impacts offsite power or lead to a station blackout event – which might occur as a result of a severe external hazard. The small safety footprint means that there is a much smaller number of potentially safety related components, which translates into a small number of components that need to be protected or robust against unscreened external hazards.

This is demonstrated in the example cases of hazard optimization of safety measures discussed above, which result in a small number of components that need to have enhanced capability against unscreened hazards. As a result, optimization of safety measures is less complicated than would be performed for an existing LWR. In the case of seismic response, enhanced capability for passive cooling using RVACS and for the vessel and internals would be the focus of optimization of safety measures, given a reliable reactor trip can be ensured.

Overall, optimization of safety measures in design is demonstrated for the PRISM plant, both in the design development phase and following site selection.

REFERENCES TO ANNEX I

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[I–2] NUCLEAR REGULATORY COMMISSION, "Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid-Metal Reactor," NUREG-1368 (1989-1994).

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ANNEX II.
ILLUSTRATIVE EXAMPLE OF THE APPLICATION OF A GRADED APPROACH TO SITE SPECIFIC PROBABILISTIC SEISMIC HAZARD ANALYSIS

II–1. INTRODUCTION

Many States have shown interest in building small modular reactors (SMRs). Therefore, there will soon be a need for site evaluation for these nuclear installations. One of the most resource intensive and time consuming aspects of site evaluation is related to the evaluation of seismic hazards. Recent examples of seismic hazard evaluation for nuclear power plants (NPPs) have shown that the time needed may be three years or longer. The question is whether such an intensive and sophisticated effort is necessary also for lower risk installations such as SMRs, or can a graded approach be used to simplify the process of seismic hazard assessment, and if so, how this can be done.

This annex provides an example for a simplified approach to the PSHA for an SMR based on the application of a graded approach. Simplification, if justified, could lead to utilize Senior Seismic Hazard Analysis Committee (SSHAC) Level 1 or preferably Level 2 with some enhancements instead of SSHAC Level 3. More details on SSHAC levels are provided in Section II–4 of this annex.

In this annex, the basis for the graded approach is explored by starting with the Fundamental Safety Principles [II–1], then considering safety requirements and then Safety Guides, including IAEA Safety Standards Series No. SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations, [II–2].

II–2. APPLICATION OF GRADED APPROACH BASED ON IAEA SAFETY STANDARDS

IAEA Safety Standards Series No. SF-1, Fundamentals Safety Principles [II–1] provides the principal basis for the application of a graded approach in order to achieve optimization of safety measures for protection of nuclear installations. Relevant excerpts from Principle 5 of Safety Fundamentals are copied below:

> **Principle 5: Optimization of Protection**
>
> **“Protection must be optimized to provide the highest level of safety that can reasonably be achieved.**
>
> 3.21. The safety measures that are applied to facilities and activities that give rise to radiation risks are considered optimized if they provide the highest level of safety that can reasonably be achieved throughout the lifetime of the facility or activity, without unduly limiting its utilization.
>
> 3.22. 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. Where there are interdependences between related actions or between their associated risks (e.g. for different stages of the lifetime of facilities and activities, for risks to different groups

> or for different steps in radioactive waste management), these must also be considered. Account also has to be taken of uncertainties in knowledge.
>
> 3.23. The optimization of protection requires judgements to be made about the relative significance of various factors, including: — The number of people (workers and the public) who may be exposed to radiation; — The likelihood of their incurring exposures; — The magnitude and distribution of radiation doses received; — Radiation risks arising from foreseeable events; — Economic, social and environmental factors. The optimization of protection also means using good practices and common sense to avoid radiation risks as far as is practical in day to day activities.
>
> 3.24. The resources devoted to safety by the licensee, and the scope and stringency of regulations and their application, have to be commensurate with the magnitude of the radiation risks and their amenability to control. Regulatory control may not be needed where this is not warranted by the magnitude of the radiation risks".

Requirements related with application of a graded approach for site evaluation are established in IAEA Safety Standards Series No. SSR-1, Site Evaluation for Nuclear Installations [II–3]. This publication provides more information on the way in which a graded approach can be applied to site evaluation for a nuclear installation.

Requirement 3 of SSR-1 [II–3] states:

> "**The scope of the site evaluation shall encompass factors relating to the site and factors relating to the interaction between the site and the installation, for all operational states and accident conditions, including accidents that could warrant emergency response actions.**"

Paragraph 4.1 of SSR-1 [II–3] states:

> "The scope of the site evaluation shall cover all external hazards, monitoring activities and site specific parameters relevant for the safety of the nuclear installation. In determining the scope of the site evaluation, a graded approach shall be applied commensurate with the radiation risk posed to people and the environment."

Paragraph 4.2 of SSR-1 [II–3] states that "The application of the safety requirements for site evaluation for nuclear installations shall be commensurate with the potential hazards associated with the nuclear installation."

Paragraph 4.3 of SSR-1 [II–3] states that "The level of detail needed in the evaluation of a site for a nuclear installation shall be commensurate with the risk associated with the nuclear installation and the site and will differ depending on the type of nuclear installation."

Paragraph 4.4 of SSR-1 [II–3] states that "The scope and level of detail of the site evaluation process necessary to support the safety demonstration for the nuclear installation shall be determined in accordance with a graded approach."

Paragraph 4.5 of SSR-1 [II–3] states:

"For site evaluation for nuclear installations other than nuclear power plants, the following shall be taken into consideration in the application of a graded approach:

(a) The amount, type and status of the radioactive inventory at the site (e.g. whether the radioactive material on the site is in solid, liquid and/or gaseous form, and whether the radioactive material is being processed in the nuclear installation or is being stored on the site);
(b) The intrinsic hazards associated with the physical and chemical processes that take place at the nuclear installation;
(c) For research reactors, the thermal power;
(d) The distribution and location of radioactive sources in the nuclear installation;
(e) The configuration and layout of installations designed for experiments, and how these might change in future;
(f) The need for active systems and/or operator actions for the prevention of accidents and for the mitigation of the consequences of accidents;
(g) The potential for on-site and off-site consequences in the event of an accident."

The above quotes indicate that a nuclear installation that has inherently lower radiological impact to the population and the environment compared to a large NPP, may have reduced scope and stringency of regulations. Therefore, the graded approach entails a proportionate reduction in the application of these requirements to a nuclear installation of lesser risk taking into account the attributes listed in Paragraph 4.5 of SSR-1 [II–3].

Most SMRs are designed and built to be safer than larger NPPs with the aim of having the flexibility of siting these reactors in areas where the criteria in terms of population density and emergency plan restrictions can be relaxed. The modular nature of an SMR provides a limit to the source term involved in a potential severe accident. However, this limit is contingent on the premise that multiple modules are not subjected to a common cause failure. This condition is an important aspect of the protection of SMRs against these common cause failures, the most probable of which would be due to external hazards affecting the whole site.

II–3. PURPOSE AND SCOPE OF THE EXAMPLE PROBABLISITC SEISMIC HAZARD ANALYSIS

This annex will focus on one of these potential sources of common cause failure; a vibratory ground motion seismic hazard. It will explore the ways in which a graded approach can be applied to seismic hazard analysis using the criteria provided in SSR-1 [II–3] and SSG-9 (Rev. 1) [II–2].

SSG-9 (Rev. 1) [II–2] provides specific recommendations related to the application of a graded approach to the evaluation of hazards due to vibratory ground motion. The example below has been chosen for a PSHA. Therefore, the following parts of this publication will only address PSHA.

II–3.1. Vibratory Ground Motion Hazard Assessment

Paragraph 9.10 of SSG-9 (Rev. 1) [II–2] states:

"The vibratory ground motion seismic hazard analysis…should be performed in accordance with the following:

(a) For the least hazardous installations, the seismic hazard input for the design may be taken from national building codes and maps.

(b) For installations in the highest hazard category, methodologies for seismic hazard assessment...(i.e. recommendations applicable to nuclear power plants) should be used.

(c) For installations categorized in the intermediate hazard category, the following approach might be applicable:

(i) If the seismic hazard assessment is typically performed using methods similar to those described in this Safety Guide, a lower seismic input for designing these installations may be adopted at the design stage, in accordance with the safety requirements for the installation;

(ii) If the database and the methods recommended in this Safety Guide are found to be disproportionately complex, time consuming and demanding in terms of the nuclear installation in question, simplified methods for seismic hazard assessment (that are based on a more restricted data set) may be used. In such cases, the seismic input finally adopted for designing these installations should be commensurate with the reduced database and the simplification of the methods, with account being taken of the fact that both factors tend to increase uncertainties."

The target for an SMR would be a categorization as described in (c) above, the choice for the PSHA would be (c) ii above where the database and the methods can be reduced and simplified but increased uncertainties need to be dealt with.

SMRs vary greatly in their design and it cannot be automatically assumed that any SMR may be categorized in the same way using the graded approach. The graded approach is to be applied step by step using the recommendations provided in SSG-9 (Rev. 1) [II–2] to decide whether the SMR in question can be categorized as described in (c) (ii) in sub-item 'ii' as mentioned above.

PSHA is the main focus of this annex. A detailed example is provided for a hypothetical NPP site where a comparison will be made between the application of a PSHA using the recommendations provided in SSG-9 (Rev. 1) [II–2] for NPPs as well as state of the art methods and, for the same site, application of a simplified PSHA.

Increased uncertainties generally imply higher hazard values, therefore, simplifications made in the methods will be preferential to conservatism.

The application of a graded approach may involve the following:

- Geological, geophysical and geotechnical (GGG) database;
- Seismological database;
- Sophistication and complexity in seismic source characterization approaches;
- Sophistication and complexity in ground motion characterization (GMC) approaches;
- Treatment of epistemic uncertainties;
- QA requirements;
- Type of independent review (e.g. participatory or end point).

The results of a graded approach applied to an SMR will also be compared to the full application of the requirements provided in SSG-9 (Rev. 1) [II–2]. The main impact is expected in the following:

- Necessary human resources;
- The time needed for the conduct of PSHA;
- Resulting hazard curves.

II–4. HISTORICAL DEVELOPMENT OF PROBABLISTIC SEISMIC HAZARD ANALYSIS

Seismic hazard analysis is based on geological, geophysical, geotechnical and seismological data sets. The database contains more and more uncertainties as its age increases. Therefore, application of strictly statistical methods using only the available database leads to unreliable results due to unquantifiable uncertainties. For this reason, it is necessary to construct models which attempt to represent the earthquake recurrence in different zones which may generate earthquakes that would affect the NPP site. These models are related to seismic source characterization. It is not sufficient to model earthquake recurrence accurately, it is also necessary to understand the characteristics of the ground motion generated by these seismic sources and their propagation to the NPP site. Ground motion exhibits great variability. For the same size earthquake and the same epicentral distance the ground motion may show significant variation which is generally characterized by the standard deviation associated with the so called GMPEs (ground-motion prediction equations; currently being referred to as ground-motion models (GMMs). Finally, site characteristics are known to modify the oncoming ground motion and this is related to the geological and geotechnical conditions at the NPP site.

Modelling that needs to be done for the seismic sources, ground motion and site response leads to a different type of uncertainty which is epistemic. In other words, given the same dataset, different experts may come up with different models. As the dataset gets larger and experts interact with each other these differences may decrease. Both the expansion of the database and greater interaction between experts require additional resources and time.

A formal methodology for integrating epistemic uncertainties within a PSHA was introduced by the SSHAC in the 1990s for a project involving all the operating NPPs in United States [II–4]. Since then, this method has been adopted and modified using experience feedback in nuclear PSHA projects. SSG-9 (Rev. 1) [II–2] also refers to this approach in a footnote.

The SSHAC approach has four levels of complexity. For nuclear projects there are examples of Level IV, Level III, Level II, and also hybrid levels (e.g. Level II for seismic source characterizations and Level III for GMC). The criteria for choosing an appropriate level for a PSHA is provided in Refs [II–4, II–5]). Two examples of a Level IV SSHAC are the Yucca Mountain project and the Swiss Nuclear project for all operating Swiss NPPs, Pegasus. The more recent of the NUREGs describes in more detail SSHAC Level III [II–6] which has found application also in several European nuclear projects. One observation that was made relates to the long duration needed even for a SSHAC Level III. Part of the reason for this stems from the fact that the original SSHAC and Pegasus were performed for operating NPPs and therefore without much schedule pressure. However, this is not the case for projects that are to be newly built.

One of the alternatives for simplifying the PSHA would be to use a lower level SSHAC to include epistemic uncertainties. This alternative is explored and presented in this annex.

In order to justify the use of simplified methods for SMRs, several important aspects of the seismic experience of NPPs need to be mentioned.

- Generally good experience for design basis exceedance of real earthquakes for NPPs. Several NPPs in the USA and Japan have experienced earthquakes where the ground motion values exceeded their design basis values. However, there was no incidence of any safety related SSC having failed because of this [II–7];
- No cliff edge effect like flooding. Due to the failure mechanisms of safety related SSCs, it is understood that global seismic failure for a NPP is a progressive process and does not present a cliff edge. This is an important attribute and one that has received much attention after the Fukushima Daiichi NPP accident. Furthermore, in SMRs many safety systems are passive. Although seismic motion will challenge these SSCs, their failure would be more progressive than sudden (compared to active components) preventing cliff edge effects.

II–5. ASSUMPTIONS IN THE EXAMPLE PROBABLISTIC SEISMIC HAZARD ANALYSIS

Following are assumptions that need to be considered in the example PSHA presented in this annex:

- The site may be affected both by faults and area sources within the region;
- The site is not within the region of influence of a major subduction zone. It is located within a region affected by shallow active crustal seismicity;
- The site is not within 100 km of a major transform fault;
- A reasonably well compiled historical and instrumental earthquake catalogue is available for the region;
- The site response characteristics can be managed through the use of appropriate GMMs. The site condition at the SMR site is represented by a single V_{S30} value and uncertainty in its measurement is disregarded for simplicity;
- The Vs-kappa adjustment to account for crustal site amplification and high frequency ground motion attenuation are disregarded;
- Fault capability and its potential interaction with vibratory ground motion is outside the scope of the example;
- Interactions between vibratory ground motion hazard and any seismically induced geotechnical hazard are ignored;
- Structural frequency range will be selected as appropriate for an SMR;
- Annual frequency of exceedance will extend to the level of 10^{-7} as such low levels of annual frequencies of exceedance (AFE) may be needed for beyond design basis evaluation and seismic PSA.

Due to the assumptions made above, and considering that the most resource intensive part of database acquisition is in the near region, it is not anticipated that the geological, geophysical and geotechnical database will be reduced. Therefore, simplification in the approaches to seismic source characterization and GMC will be the main focus of the example.

II–6. PROBABLISTIC SEISMIC HAZARD ANALYSIS TECHNICAL TEAM AND ITS STRUCTURE

The PSHA team is established from a technical group consisting of seismic hazard experts in different components of PSHA modelling and analysis as well as a review team to assess the technical quality of the study. The review team has to also assess the physical soundness of PSHA results that will be used in design and/or probabilistic safety analysis of SMRs. The review has also to consider the software used in the PSHA modelling and computation of PSHA results. To this end, studies such as in Ref. [II–8] need to be considered to set the essential criteria required and expected from a reliable PSHA software.

The team is coordinated by an integrator who has a good understanding of seismic source, ground motion, and PSHA (logic-tree framework) modelling. The integrator as well as the seismic source characterization and GMC leads have to be familiar with the relevant IAEA safety standards, in particular SSG-9 (Rev. 1) [II–2]. A source modeller with a strong background in earthquake catalogue compilation, completeness and de-clustering analyses, stochastic recurrence models, and source modelling of diffuse seismicity, gridded seismicity, smoothed seismicity, and faults is expected to lead the seismic source characterization component of PSHA. The seismic source characterization lead has to be familiar with the logic-tree framework. Depending on the complexity of the seismo-tectonics of the region, the seismic source characterization lead can establish a working group that can also include local/regional experts from the site specific seismic source characterization model.

The ground motion modeller (the GMC lead) is expected to be up-to-date in ground motion models developed for different seismotectonic regions (such as subduction zones, shallow active crustal regions, stable continental regions, volcanic regions, etc.). The GMC leads also need to have a good understanding in modelling methodologies such as backbone modelling through scaling, adjustment and/or Sammon's map (a high dimensional visualization tool to assess the differences, for example, in median spectral amplitude estimations of multiple GMMs under different predictors such as distance, magnitude etc., see Ref. [II–9]. Knowing these methodologies will indicate that the ground motion modeller has expertise in ground motion logic tree framework. It would also be preferable if the GMC lead has skills in ground motion model development. As in the case of seismic source characterization lead, the GMC lead can also establish a working group for developing the GMC model. It is preferable that the integrator can also assume one of the seismic source characterization or GMC lead roles depending on the level of his/her expertise and knowledge in these topics. A knowledgeable and experienced integrator in seismic source characterization and GMC modelling would increase the interaction between the seismic source characterization and GMC teams that essentially shortens the integration time of seismic source characterization and GMC models. The seismic source characterization and GMC teams are expected to meet regularly during the modelling phase. The seismic source characterization and GMC leads need to have continuous interaction.

The PSHA modeller runs the site specific seismic source characterization and GMC models by using a PSHA computational platform that is approved by the integrator, seismic source characterization and GMC leads as well as the reviewers. The PSHA modeller reports to the integrator but also has continuous interaction with the seismic source characterization and GMC leads about the respective models. The PSHA modeller is expected to know comprehensive modelling techniques for the seismic sources of concern in the project and be able to run sensitivity analyses, such as Tornado diagrams and analysis of variance, needed for understanding the most important seismic parameters in seismic source characterization and

GMC and their overall contribution to the epistemic uncertainty. The PSHA modeller is to also be capable of running deaggregation analysis to have a clear vision in the most contributing (critical) earthquake scenarios for given spectral ordinates and annual frequencies of exceedance.

The review team consists of at least two experts each having significant knowledge in seismic source characterization and GMC, respectively. The review team is involved in the entire process in order to not misjudge (or misinterpret) some of the critical modelling decisions in seismic source characterization and GMC. The reviewers need to perform a top-to-bottom review at the end of the project to endorse the successful termination of the study with goals reached. This phase can be realized by setting concurrent meetings where the integrator, the leads, and the seismic source characterization and GMC teams make detailed presentations for technical discussions about the results or models used for producing the results. Figure II–1 shows the team structure and the interaction within the team structure. All group/member interactions are bidirectional and intersect at the integrator as the technical responsible of the entire study. The integrator may have an assistant who helps him/her in documenting the project meetings, the reports, interaction with the technical teams and the teams of the regulatory body and the client. The assistant is not expected to have technical knowledge on the PSHA but have strong communication skills and document archiving, which includes the technical documents (scientific/topical papers and reports) and meeting minutes, technical presentations, interim project products, etc.

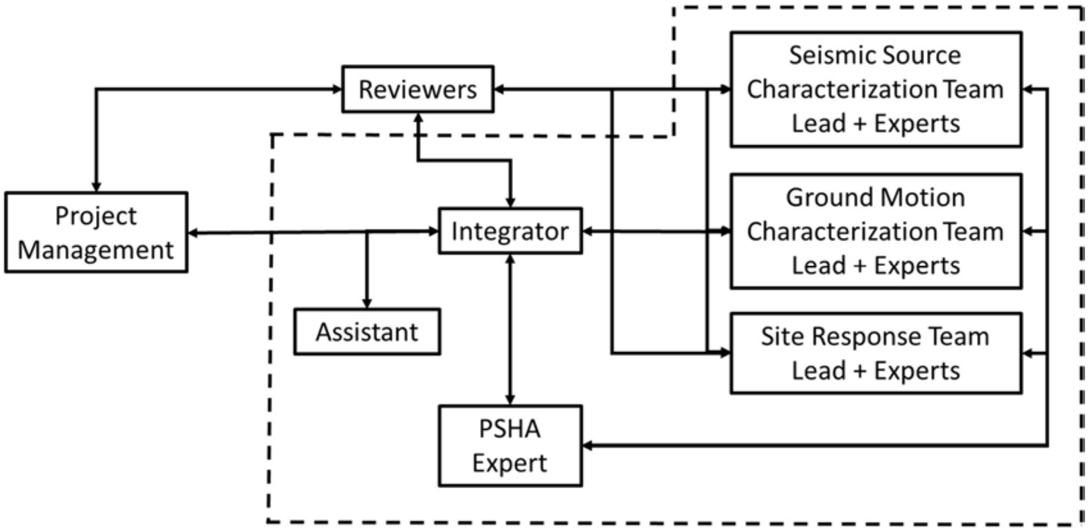


FIG. II–1. Team structure of the site specific PSHA of an SMR and the interaction between the groups.

II–7. GRADED PROBABLISTIC SEISMIC HAZARD ANALYSIS METHODOLOGY

The PSHA study aims to compute the ground motion intensity measure levels for the SMR site at the target annual frequencies of exceedance recommended in SSG-9 (Rev. 1) [II–2]. This objective is accomplished by considering the epistemic uncertainty in seismic source characterization and GMC through model evaluation and integrating it through mutually exclusive and collectively exhaustive (MECE) PSHA logic tree framework. The mutual exclusiveness and collective exhaustiveness are needed for the mathematically sound and unbiased distribution of hazard curves. The epistemic uncertainty in the seismic source characterization and GMC model parameters need to account for insufficient knowledge in the local seismic data, the relevant models compiled from literature, and the relevant methods used in the computation of various seismic source characterization and GMC components. These can be referred to as ‘existing information’. The distribution of epistemic uncertainty is then evaluated through (a) best estimates from existing information (centre), (b) interpretations yielding alternatives to best estimates (body), and (c) physically plausible limits of existing information that are yet to occur (range). The distribution of epistemic uncertainty established by this systematic approach is referred to as “the technically defensible interpretations (TDI) of the centre, the body and the range (CBR)” [II–5, II–6]. It is then endorsed by the reviewer team for seismic source characterization and GMC models.

The graded PSHA methodology aims to reach the above objective within a region encapsulating the seismicity that significantly contributes to the SMR’s seismic hazard. The methodology tailors the CBR of critical seismic parameters and relevant seismic models by focusing on this region. The simplifications are not to yield inaccurate or indefensible PSHA results and need to be on the conservative side. To this end, the technical decisions taken by the experts involved are to be justified at every step in the methodology. The following lines explain the graded PSHA methodology step by step.

Step 0: Assess the existing geological, geotechnical and geophysical information for the study region for a proposed SMR site

This step provides input for Step #1 (preliminary PSHA study). The technical team compiles available information in a seismotectonic database and identifies seismic sources in the region to propose alternative models for preliminary PSHA. The technical team can consider reliable previous PSHA studies at this step for preliminary source and ground motion characterization of the preliminary PSHA. Such compilation is done from peer-reviewed papers or regional hazard projects. If, for example, the technical team concludes that the GMMs determined from previous PSHA studies are not up to date, the GMC team can propose another set of GMMs (from literature) representing the seismotectonic settings featured by the seismic sources that are included in the preliminary PSHA model. Note that the information gathered at this step is also useful for the final seismic source characterization, GMC and PSHA models.

Step 1: Perform a preliminary PSHA at the site of interest

Step 1 is performed taking into consideration the recommendations of SSG-9 (Rev. 1) [II–2], i.e., the size of the region to be investigated when assessing vibratory ground motion hazards is large enough to incorporate all seismogenic structures that could affect the nuclear installation. The extent of this region is typically a few hundred kilometres in radius from site, or in keeping with the national requirements of the State.

This step is performed to delineate the seismic sources (faults, areal sources, etc.) that significantly contribute to the vibratory ground motion hazard at the SMR site. Step 0 provides the input data for this step. The deliverable of Step 1 is the seismic sources as well as their boundaries that will be used in the seismic source characterization model. This is explained in the following paragraphs.

The detection of most contributing seismic sources and the corresponding earthquake scenarios in terms of magnitude (M)-distance (R) pairs are done by deaggregation analysis at the Peak Ground Acceleration (PGA) and other discrete periods covering $T \leq 1.0$ s range. This period interval is chosen deliberately as spectral ordinates within this period range as well as PGA are used in the SMR’s SSC design. The deaggregation analyses are performed for AFEs of 10^{-4}, 10^{-5}, and 10^{-6} that are imported for SMR’s SSC design as well as probabilistic safety analysis. The graded approach considers the seismic sources contributing to 90% of the total mean hazard in the PSHA model. These seismic sources are referred to as controlling seismic sources and their modal magnitudes can guide the seismic source characterization and GMC teams for the critical maximum magnitudes to be considered in their models. The source-to-site distance (R) determined from deaggregation analysis would draw the geographical boundaries to delineate the geometries of controlling seismic sources in the PSHA model.

The preliminary PSHA phase can consider reliable previous PSHA studies for the region where an SMR will be sited. The previous PSHA studies can be compiled from peer-reviewed papers or regional hazard projects. The model used in these studies does not need to be sophisticated but it has to be evaluated by the technical team to confirm that its technical quality satisfies the minimum expected standards from a PSHA study. It will only be used for determining the areal bound of the actual PSHA study and the technical team can complete any missing (or unsatisfactory) components of the model considering the overall seismo-tectonic features of the region of interest. For example, if the preliminary PSHA model does not include recent ground-motion models, the GMC team can propose a set of GMMs representative of the seismotectonic settings featured by each seismic source in the preliminary PSHA model. In the preliminary PSHA phase, the conditions at the SMR site can be represented by the site information gathered from available geological and geophysical site measurements. This strategy would not jeopardize the reliability of the determination of controlling sources as well as the geographical boundary to delineate the geometry of controlling seismic sources that are considered in the actual PSHA model.

Step 2: Seismic source characterization model

The seismic source characterization lead and the experts collaborating in source modelling conduct an enhanced literature study and gather as much seismic data as possible for the controlling areal and fault sources whose geometries are determined in Step 1 (Preliminary PSHA). The literature survey needs to compile reliable (peer-reviewed) local and/or international references as the seismic source characterization model established from such references would be more defendable. Step 0 can provide a valuable background for such an input.

Considering the above discussions, the essential inputs of this step are as follows:

- The earthquake catalogue;
- The boundary conditions (leaky vs. non-leaky) for the propagation of fictitious ruptures in controlling areal sources;
- The styles of faulting in controlling sources;
- The seismogenic depths;
- The determination of maximum magnitudes of controlling sources;
- The magnitude-area scaling models;
- The recurrence models;
- The depth to top-of-rupture.

The deliverable of Step 2 is the source characterization model and the seismic source characterization logic tree addressing the CBR in the seismic source characterization components.

The seismic source characterization model is developed under the guidance of preliminary PSHA deaggregation results and the epistemic uncertainty of model parameters used for describing the seismicity of controlling areal and fault sources needs to be carefully evaluated. The exhaustive alternatives in some of the source model parameters can unnecessarily complicate the seismic source characterization logic tree structure during the integration phase without a significant improvement in the PSHA results. To this end, the interaction between the seismic source characterization lead, the seismic source characterization team and the technical integrator needs to be intensive to develop a well-balanced seismic source characterization model that utilizes the most critical (and necessary) seismic parameters. The associated seismic source characterization logic-tree framework will properly portray the epistemic uncertainty in each one of these parameters for a physically defendable seismic source characterization model.

The earthquake catalogue covers the modal magnitude(s) from deaggregation of preliminary PSHA results to envisage the maximum magnitude (M_{max}) range of the seismic source(s). The robustness of the recurrence models and the associated activity rates depend on the earthquake catalogue. If the earthquake catalogue is evaluated to be insufficient, the recurrence models and their parameters need to be imported from the analogues of tectonic regimes. The seismic source characterization lead and the source modelling team provide physical evidence on the usability of the stochastic recurrence parameters compiled from the analogue regions in case there are intolerable deficiencies in the earthquake catalogue. This evaluation and its justification need to be documented clearly.

Step 3: GMC model

The GMC model of the graded PSHA methodology is established from backbone modelling approach [II–10] for median PGA and spectral amplitude estimation. The backbone model inherently evaluates the epistemic uncertainty in median ground-motion estimations by using a backbone ground motion model (GMMBB) and populates the GMC logic tree with the scaled and/or adjusted versions of GMMBB to capture the epistemic uncertainty in median ground motions. The scaling and/or adjustment represents the specific features and uncertainties involved in sources, path, and site effects of the spectral amplitudes defining the earthquake induced demand at the SMR site. The epistemic uncertainty range described in this manner is, in fact, a probability distribution and the logic tree branches, established from the scaled backbone, with proper weights can represent this distribution. This point is explained further and is clarified in the example PSHA case study. The conventional multiple-GMM approach that populates the GMC logic trees by assigning weights to the candidate GMMs identified through some statistical testing criteria may fail to cover the CBR in median spectral amplitudes for specific magnitude-distance ranges.

The main inputs of the GMC model are the candidate GMMs used in the backbone modelling, which is elaborated further in the below lines. The candidate GMMs are also used in modelling the aleatory variability (standard deviation or the so-called sigma), which is the integral component of GMC. The deliverable of Step 3 is the ground motion model for median spectral amplitudes associated with the standard deviation as well as the GMC logic tree covering the CBR in these two components. The starting phase of the GMC model is to identify candidate GMMs by considering the controlling seismic source(s) identified in Step 1. The identification of candidate GMMs bears on some well recognized rules by the technical PSHA community such as those put forward in Ref. [II–11] as well as the requirements specific to the project. The below bullets give a few exemplary rules used in screening out candidate GMMs:

- The GMM is from an irrelevant tectonic regime;
- The GMM is not peer-reviewed;
- The GMM has been superseded;
- The range of applicability of the GMM is narrow (i.e., the distance and magnitude ranges of the model cannot cover the geographic boundaries determined in Step 1);

- The GMM uses inappropriate definitions for explanatory variables, such as local magnitude (M_L) instead of moment magnitude (M_w) or epicentral distance (R_{EPI}) instead of one of the extended source distance metrics such as Joyner-Boore distance (R_{JB}) or rupture distance (R_{RUP}).

The GMC lead and the GMC team can establish the rules to select the candidate GMMs by considering the criteria given above, the ones proposed in the relevant literature, and the specific limitations imposed by the PSHA project. As the backbone modelling approach yields the distribution of CBR for median spectral amplitudes, the GMC logic tree framework uses this distribution and discretizes it with proper weights at specific percentiles of the distribution. The PSHA example explains this point further.

Step 4: Integration and Documentation

After conducting the evaluation of 'existing information' for seismic source characterization and GMC models, the technical integrator and the seismic source characterization and GMC leads integrate the seismic source characterization and GMC models as a full PSHA model under the complete logic tree framework. The objective of the integration phase is to ensure that CBR is captured with justifiable logic tree weights. Justification for the technical bases for the weights on different models in the final logic tree and consequently in hazard distribution, as well as the exclusion of the models and methods proposed, needs to be documented. A clear documentation of the entire project work would increase its chances to be endorsed by review team. FIG. II–2 depicts the workflow of the project steps.

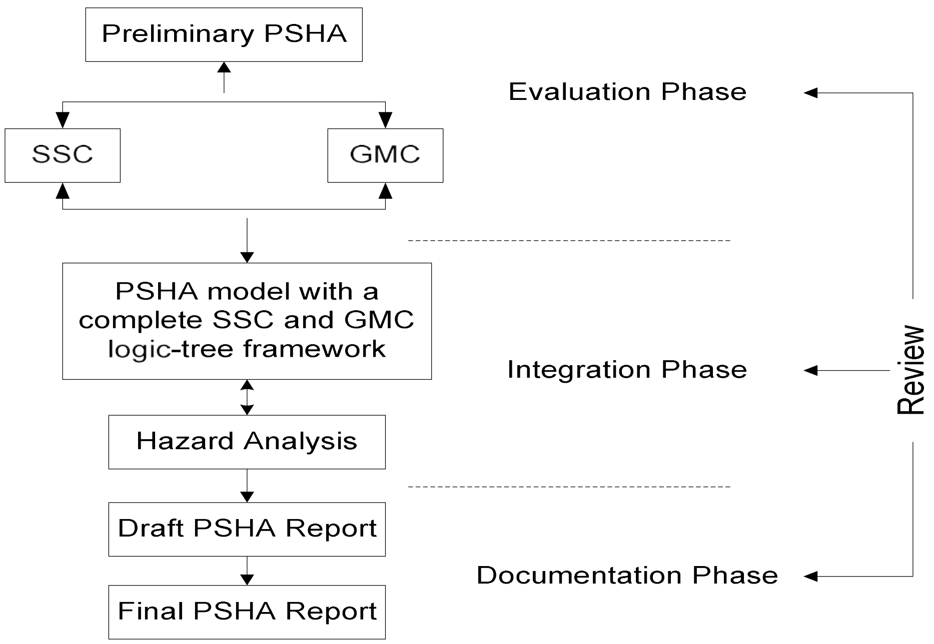


FIG. II- 2. Illustration of workflow (evaluation, integration and documentation phases) for the graded PSHA methodology for SMRs site

The documentation phase gives the details of seismic source characterization, GMC and PSHA models, and includes the sensitivity analyses to report the level of contribution of each model parameter to the overall epistemic uncertainty in PSHA. This systematic is the unbiased and technically defendable way of addressing the final PSHA results.

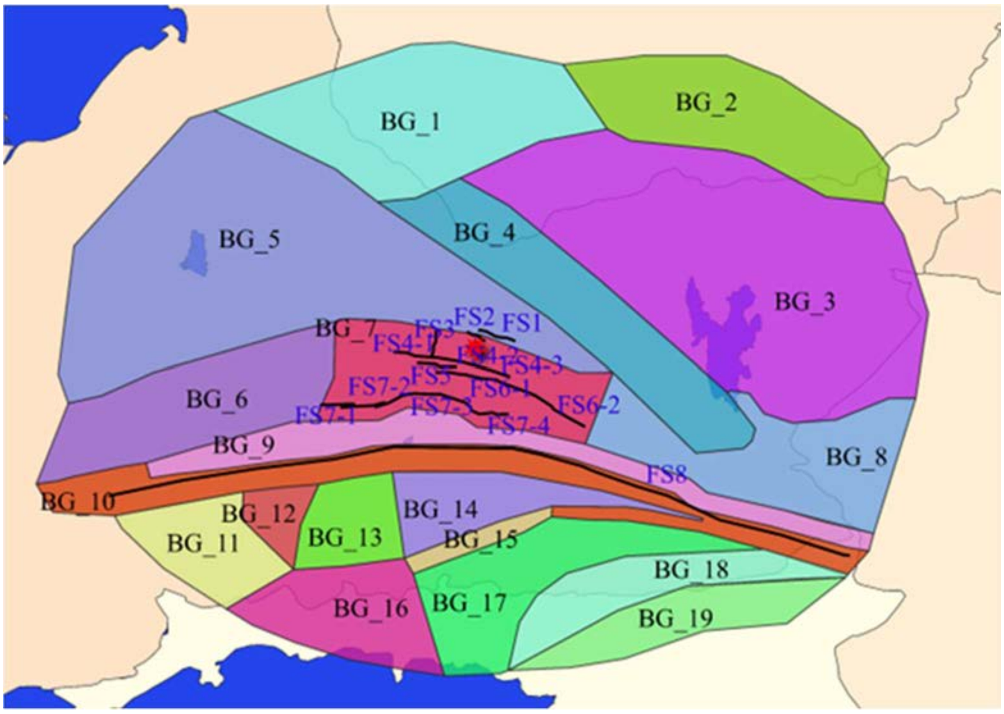
II–8. EXAMPLE PROBABILISTIC SEISMIC HAZARD ANALYSIS

For the example PSHA, the SMR site considered is under the effect of shallow active crustal seismicity and it is surrounded by active seismic sources of moderate to high seismicity. The following subsections describe the step-by-step graded PSHA methodology for the example PSHA case.

II–8.1. Preliminary Probabilistic Seismic Hazard Analysis Model

The preliminary model considers the seismic sources encompassing a 300 km radius around the SMR site (as recommended in SSG-9 (Rev. 1) [II–2]) that are compiled from existing comprehensive regional studies. FIG. II–3 displays the modelled fault (FIG. II–3. a) and areal seismic sources (FIG. II–3. b) in this context. The fault model also considers the background seismicity represented by different colour schemes.

(a)



(b)

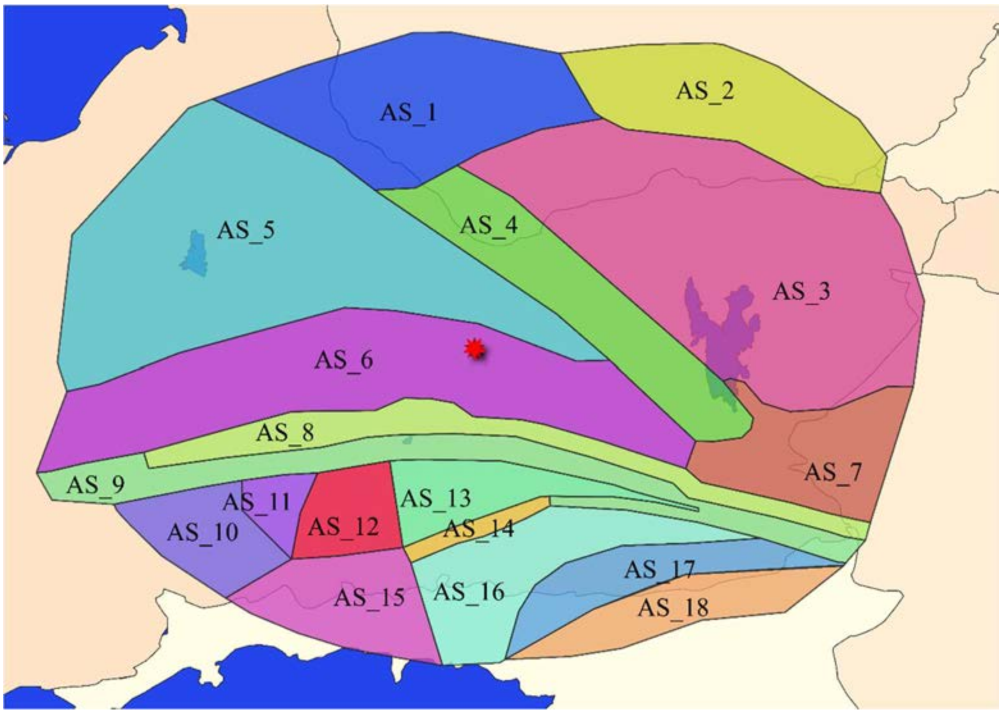


FIG. II–3. The location of SMR site: (a) Fault sources and background seismicity, (b) Areal sources.

The seismic source characterization team incorporates eighteen areal sources in the areal source model by considering differences in regional seismo-tectonics, seismic activity, and orientations as well as faulting styles of diffuse faults obtained from fault-plane solutions. This information is gathered from the regional earthquake catalogues. Table II–1 lists the main hazard input parameters of areal source models. Since the preliminary PSHA is only for guidance to final PSHA model, the epistemic uncertainty is considered in a very limited way. The seismic source characterization logic tree of areal source model only accounts for the style of faulting (SoF) uncertainty that enters into calculations with equal weights for each alternative SoF. The variations in strike angles of diffuse faults and hypocentral depths are considered as part of aleatory random variability with a uniform distribution as given in Table II–1. The seismic source characterization team uses the truncated Gutenberg-Richter magnitude recurrence model as the representative of earthquake occurrence in all areal sources.

TABLE II–1. HAZARD INPUT PARAMETERS OF AREAL SOURCES FOR PRELIMINARY ANALYSIS

Source Name	Upper/ lower seismogenic crust depth (km)	Hypocentral depth (km) (Uniform)(1)	SoF(2)	Strike (degree) (Uniform)(1)	Dip (degree)	Magnitude Recurrence Parameters (order: a, b, M_{max})(3)
AS_1	0.1 / 45	btw 5 and 41; Δ = 4.0	R	btw 90 and 120; Δ = 10	40	3.22, 0.95, 7.45
			SS	btw 90 and 120; Δ = 10	90	
AS_2	0.1 / 45	btw 5 and 41; Δ = 4.0	R	btw 60 and 90; Δ = 10	40	3.84, 1.01, 7.45
			SS	btw 60 and 90; Δ = 10	90	
AS_3	0.1 / 15	btw 3 and 15; Δ = 3.0	R	btw 110 and 150; Δ = 10	40	2.76, 0.92, 6.5
			N	btw 110 and 150; Δ = 10	60	
			SS	btw 20 and 60; Δ = 10	90	
AS_4	0.1 / 15	btw 3 and 15; Δ = 3.0	R	btw 110 and 150; Δ = 10	40	2.66, 0.91, 6.5
			N	btw 110 and 150; Δ = 10	60	
			SS	btw 20 and 60; Δ = 10	90	
AS_5	0.1 / 15	btw 3 and 15; Δ = 3.0	R	btw 110 and 150; Δ = 10	40	2.95, 0.93, 6.5
			N	btw 110 and 150; Δ = 10	60	
			SS	btw 20 and 60; Δ = 10	90	
AS_6	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 70 and 110; Δ = 10	40	3.95, 1.02, 7.2
			N	btw 30 and 210; Δ = 10	60	
			SS	btw 330 and 390; Δ = 10	90	
AS_7	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 70 and 120; Δ = 10	40	3.66, 1.07, 7.2
AS_8	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 40 and 120; Δ = 10	40	3.78, 0.99, 7.25
			SS	btw 40 and 120; Δ = 10	90	
AS_9	0.1 / 15	btw 3 and 15; Δ = 3.0	R	btw 330 and 380; Δ = 10	40	2.67, 0.66, 8.3
			N	btw 330 and 380; Δ = 10	60	
			SS	btw 60 and 110; Δ = 10	90	
AS_10	0.1 / 20	btw 3 and 18; Δ = 3.0	N	btw 80 and 100; Δ = 10	60	3.04, 0.92, 7.25
			SS	btw 80 and 100; Δ = 10	90	
AS_11	0.1 / 20	btw 3 and 18; Δ = 3.0	N	btw 90 and 110; Δ = 10	60	2.84, 0.88, 7.25
			SS	btw 180 and 200; Δ = 10	90	
AS_12	0.1 / 16	btw 3 and 15; Δ = 3.0	R	btw 180 and 200; Δ = 10	40	3.05, 0.85, 7.25
			SS	btw 90 and 110; Δ = 10	90	
AS_13	0.1 / 20	Btw 3 and	R	btw 340 and 380; Δ = 10	40	2.96, 0.81, 7.65

TABLE II–1. HAZARD INPUT PARAMETERS OF AREAL SOURCES FOR PRELIMINARY ANALYSIS (Cont'd)

Source Name	Upper/ lower seismogenic crust depth (km)	Hypocentral depth (km) (Uniform)[(1)]	SoF[(2)]	Strike (degree) (Uniform)[(1)]	Dip (degree)	Magnitude Recurrence Parameters (order: a, b, M_{max})[(3)]
		18;	N	btw 340 and 380; Δ = 10	60	
		Δ = 3.0	SS	btw 70 and 110; Δ = 10	90	
AS_14	0.1 / 20	btw 3 and 18; Δ = 3.0	SS	btw 50 and 70; Δ = 10	90	2.31, 0.82, 7.65
AS_15	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 40 and 60; Δ = 10	40	3.05, 0.87, 7.25
			N	0	60	
			SS	btw 60 and 80; Δ = 10	90	
AS_16	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 335 and 355; Δ = 10	40	2.59, 0.8, 7.65
			N	btw 335 and 355; Δ = 10	60	
			SS	btw 65 and 85; Δ = 10	90	
AS_17	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 310 and 330; Δ = 10	40	2.68, 0.83, 7.65
			N	btw 310 and 330; Δ = 10	60	
			SS	btw 40 and 60; Δ = 10	90	
AS_18	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 335 and 355; Δ = 10	40	3.14, 0.87, 7.65
			N	btw 335 and 355; Δ = 10	60	
			SS	btw 60 and 80; Δ = 10	90	

(1) Δ: either equal depth or strike angle increment with uniform distribution
(2) SoF: style of faulting; R: Reverse; SS: Strike-slip; N: Normal
(3) a: number of earthquakes with $M_w \geq 0$; $b = \beta/\ln(10)$; M_{max}: in moment magnitude, M_w

Tables II–2 and II–3 present the seismic source input parameters of fault and background seismicity considered in the fault model, respectively. As in the case of areal source model, the seismic source characterization team compiles the relevant information from earthquake catalogues and the epistemic uncertainty is only considered in SoFs (background seismicity) in the same way as explained for the area source model.

The seismic source characterization team members select the characteristic recurrence model from Ref. [II–12] for the fault model, which is frequently used to represent the earthquake occurrences for this type of source modelling. The M_{min} values defined in Table II–2 correspond to the minimum magnitudes defined for fault sources. However, the background seismicity which is the complementary seismic source of the fault source model has M_{min} = 5.0. Accordingly, the M_{min} defined for fault sources are not be confused with the minimum magnitude of the PSHA integral.

TABLE II–2. HAZARD INPUT PARAMETERS OF FAULT SOURCES IN THE PRELIMINARY FAULT MODEL

Source Name	Upper/ lower seismogenic crust depth (km)	SoF	Dip (degree)	M_{min} (fault)	b value	M_{char}	M_0 (Nm)
FS1	1 / 25	R	90	6	1.04	6.4	8.81E+14
FS2	1 / 25	N	60	5.75	1.04	6.1	3.26E+14
FS3	1 / 25	N	60	6	1.04	6.4	9.19E+14
FS4-1	1 / 25	R	40	6	1.04	6.8	4.98E+15
FS4-2	1 / 25	R	40	6	1.04	6.4	2.02E+15
FS4-3	1 / 25	R	40	5.85	1.04	6.2	1.29E+15
FS5	1 / 25	R	40	6	1.04	6.5	3.16E+15
FS6-1	1 / 25	R	40	6	1.04	6.8	9.96E+15
FS6-2	1 / 25	R	40	6	1.04	7.1	1.96E+16
FS7-1	1 / 25	R	40	6	1.04	6.4	3.63E+15
FS7-2	1 / 25	R	40	6	1.04	6.6	5.71E+15
FS7-3	1 / 25	R	40	6	1.04	7.5	4.35E+16
FS7-4	1 / 25	R	40	6	1.04	6.5	4.55E+15
FS8	0 / 25	SS	90	6.5	0.661	8.3	1.41E+19

TABLE II–3. HAZARD INPUT PARAMETERS OF BACKGROUND SOURCES IN THE FAULT MODEL PRELIMINARY ANALYSIS

Source Name	Upper/ lower seismogenic crust depth (km)	Hypocentral depth (km) (Uniform)[1]	SoF[2]	Strike (degree) (Uniform)[1]	Dip (degree)	Magnitude Recurrence Parameters (order: a, b, M_{max})[3]
BG_1	0.1 / 45	btw 5 and 41; Δ = 4.0	R	btw 90 and 120; Δ = 10	40	3.22, 0.95, 7.45
			SS	btw 90 and 120; Δ = 10	90	
BG_2	0.1 / 45	btw 5 and 41; Δ = 4.0	R	btw 60 and 90; Δ = 10	40	3.84, 1.01, 7.45
			SS	btw 60 and 90; Δ = 10	90	
BG_3	0.1 / 15	btw 3 and 15; Δ = 3.0	R	btw 110 and 150; Δ = 10	40	2.76, 0.92, 6.5
			N	btw 110 and 150; Δ = 10	60	
			SS	btw 20 and 60; Δ = 10	90	
BG_4	0.1 / 15	btw 3 and 15; Δ = 3.0	R	btw 110 and 150; Δ = 10	40	2.66, 0.91, 6.5
			N	btw 110 and 150; Δ = 10	60	
			SS	btw 20 and 60; Δ = 10	90	
BG_5	0.1 / 15	btw 3 and 15; Δ = 3.0	R	btw 110 and 150; Δ = 10	40	2.95, 0.93, 6.5
			N	btw 110 and 150; Δ = 10	60	
			SS	btw 20 and 60; Δ = 10	90	
BG_6	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 50 and 90; Δ = 10	40	3.95, 1.06, 7.2
			SS	btw 50 and 90; Δ = 10	90	
BG_7	0.1 / 25	btw 3 and 21; Δ = 3.0	R	btw 70 and 110; Δ = 10	40	3.454, 1.04, 6.0
			N	btw 30 and 210; Δ = 10	60	

TABLE II–3. HAZARD INPUT PARAMETERS OF BACKGROUND SOURCES IN THE FAULT MODEL PRELIMINARY ANALYSIS (Cont'd)

Source Name	Upper/ lower seismogenic crust depth (km)	Hypocentral depth (km) (Uniform)(1)	SoF(2)	Strike (degree) (Uniform)(1)	Dip (degree)	Magnitude Recurrence Parameters (order: a, b, M_{max})(3)
BG_8	0.1 / 20	btw 3 and 18; Δ = 3.0	SS	btw 330 and 390; Δ = 10	90	3.77, 1.07, 7.2
			R	btw 70 and 120; Δ = 10	40	
BG_9	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 40 and 120; Δ = 10	40	3.78, 0.99, 7.25
			SS	btw 40 and 120; Δ = 10	90	
BG_10	0.1 / 15	btw 2 and 14; Δ = 3.0	R	btw 330 and 380; Δ = 10	40	2.67, 0.66, 8.3
			N	btw 330 and 380; Δ = 10	60	
			SS	btw 60 and 110; Δ = 10	90	
BG_11	0.1 / 20	btw 3 and 18; Δ = 3.0	N	btw 80 and 100; Δ = 10	60	3.04, 0.92, 7.25
			SS	btw 80 and 100; Δ = 10	90	
BG_12	0.1 / 20	btw 3 and 18; Δ = 3.0	N	btw 90 and 110; Δ = 10	60	2.84, 0.88, 7.25
			SS	btw 180 and 200; Δ = 10	90	
BG_13	0.1 / 16	btw 3 and 15; Δ = 3.0	R	btw 180 and 200; Δ = 10	40	3.05, 0.85, 7.25
			SS	btw 90 and 110; Δ = 10	90	
BG_14	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 340 and 380; Δ = 10	40	2.96, 0.81, 7.65
			N	btw 340 and 380; Δ = 10	60	
			SS	btw 70 and 110; Δ = 10	90	
BG_15	0.1 / 20	btw 3 and 18; Δ = 3.0	SS	btw 50 and 70; Δ = 10	90	2.31, 0.82, 7.65
BG_16	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 40 and 60; Δ = 10	40	3.05, 0.87, 7.25
			N	0	60	
			SS	btw 60 and 80; Δ = 10	90	
BG_17	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 335 and 355; Δ = 10	40	2.59, 0.8, 7.65
			N	btw 335 and 355; Δ = 10	60	
			SS	btw 65 and 85; Δ = 10	90	
BG_18	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 310 and 330; Δ = 10	40	2.68, 0.83, 7.65
			N	btw 310 and 330; Δ = 10	60	
			SS	btw 40 and 60; Δ = 10	90	
BG_19	0.1 / 20	btw 3 and 18; Δ = 3.0	R	btw 335 and 355; Δ = 10	40	3.14, 0.87, 7.65
			N	btw 335 and 355; Δ = 10	60	
			SS	btw 60 and 80; Δ = 10	90	

(1) Δ: either equal depth or strike angle increment with uniform distribution
(2) SoF: style of faulting; R: Reverse; SS: Strike-slip; N: Normal
(3) a: number of earthquakes with $M_w \geq 0$; $b = \beta/\ln(10)$; M_{max}: in moment magnitude, M_w

The seismic source characterization team uses the magnitude-rupture area scaling relationship from Ref. [II–13] and minimum magnitude of the preliminary PSHA integral is taken as M_w =

5. These decisions by the seismic source characterization team members are unanimously taken since they are considered as valid and technically defensible for preliminary PSHA. M_w 5 is based on SSG-9 (Rev. 1) [II–2] and considered to be appropriate for a new nuclear installation. As for the ground-motion model, the GMC team selects the model from Ref. [II–14] for preliminary PSHA. This model is one of the most complete GMMs in the current literature for shallow active crustal seismicity and it is implemented extensively in various site specific PSHA studies (e.g. Refs [II–15–II–17]). At this point, the site conditions at the foundation level of SMR site is represented by V_{S30} = 900 m/s that is determined from the in-situ site exploration. Therefore, the GMC team also uses V_{S30} = 900 m/s in their model that is presented in the latter parts of this document.

The preliminary PSHA is performed for area and fault source models separately to observe the behaviour of their contributions to hazard at the spectral periods and AFEs of interest. This is achieved by deaggregation analysis as explained in the first step of graded approach at T = 0.1 s, 0.3 s, 0.5 s, and 1.0 s, and AFEs of 10^{-4}, 10^{-5}, and 10^{-6}. FIG. II–4 shows the representative deaggregation results obtained from the fault source model at short (represented by T = 0.1 s) and intermediate (represented by T = 1.0 s) periods at AFE = 10^{-4}. The deaggregation results are presented as vertical column bars representing the percent contributions in different magnitude-distance bins (i.e. earthquake scenarios) to the mean hazard level at this specific AFE. The magnitude and distance bin widths are selected as 0.2 units and 5 km, respectively, for magnitude and distances ranging between $5.0 \leq M_w \leq 8.0$ and $0 \leq R_{RUP} \leq 300$ km, respectively. Sample deaggregation results in this figure suggest that the fault sources within the first 50 km contribute to the mean hazard at the SMR site for short and intermediate periods. This topic is discussed further in the following paragraphs.

a)

b)

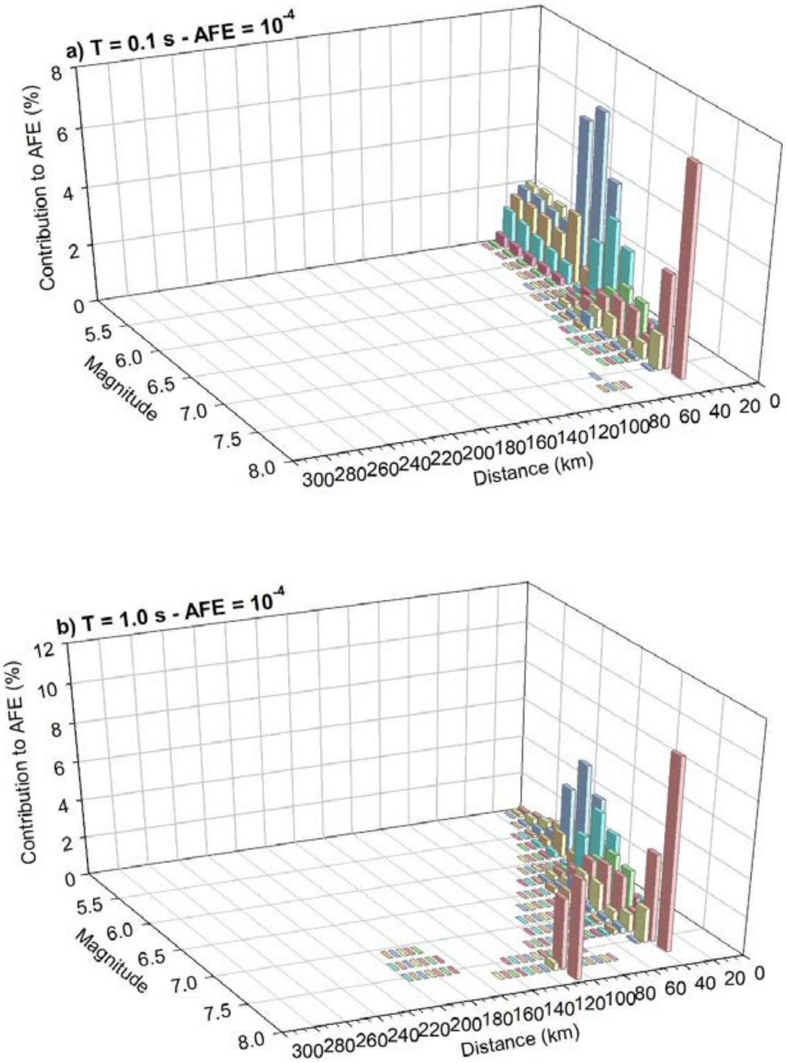


FIG. II–4. Representative deaggregation plots of fault source model for AFE = 10^{-4}: a) T = 0.1 s, b) T = 1.0 s

The deaggregation results are evaluated for assessing the contributions of seismic sources to the total mean hazard computed for area and fault source models. Given a source model (fault or area source model), the distance-wise variation in cumulative contributions of sources to the mean hazard is used by the TI and seismic source characterization lead to decide on the cut-off distance for delineating the boundary around the SMR site. The cut-off distance will pinpoint the seismic sources to be considered in the final PSHA model as explained in Step 1.

Figs II–5 and II–6 show the cumulative contributions of seismic sources for area and fault source models, respectively. Each panel in these figures represent a different period and different line colours in the panels show different AFE levels. The plots suggest that different source modelling (area vs. fault), differences in AFE levels, and different spectral period ranges affect the seismic source contributions. However, following the “90% contribution to total hazard” rule in Step 1, the TI and seismic source characterization lead decide to consider the first 50 km from the SMR site for final PSHA model as the aforementioned effects collectively lose their significances over the mean hazards of both area and fault source models. (Sources located at distances greater than 50 km have very limited effects on the vibratory ground motion hazard at the SMR site).

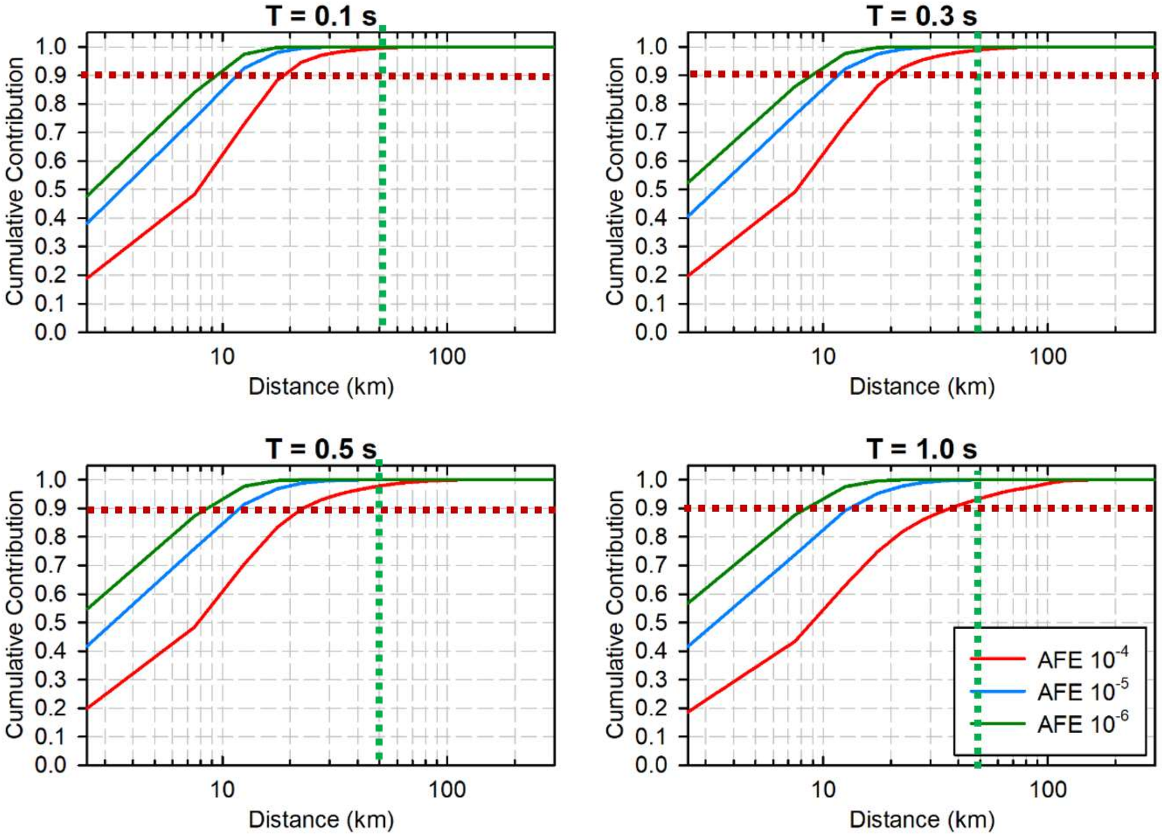


FIG. II–5. Distance-dependent cumulative contributions of sources to the total hazard in area source model. Red dashed lines show the 90% contribution, green dashed lines show the 50 km cut-off used in the PSHA model (seismic source characterization and GMC modelling and logic-tree).

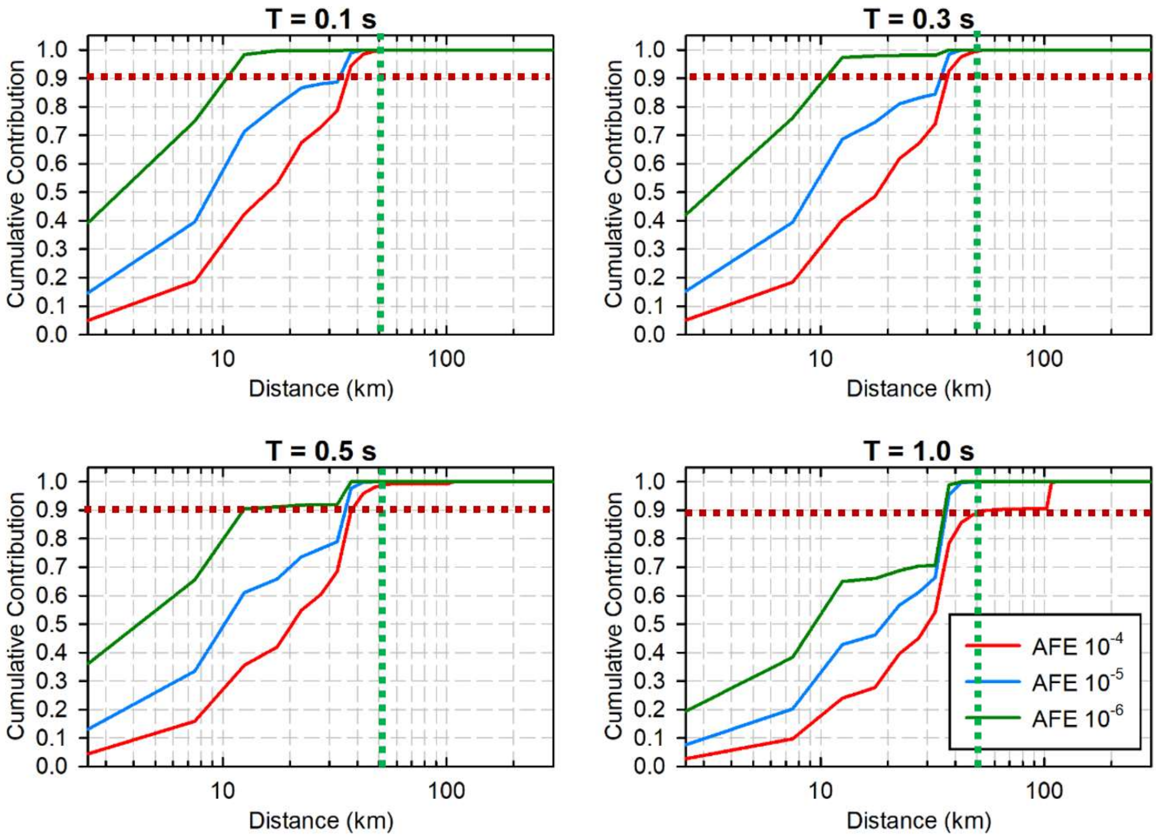


FIG. II–6. Distance-dependent cumulative contributions of sources to the total hazard in fault source model. Red dashed lines show the 90% contribution, green dashed lines show the 50 km cut-off used in the PSHA model (seismic source characterization and GMC modelling and logic-tree).

II–8.2. Ground Motion Characterization Model

The ground-motion characterization for median spectral amplitudes is based on the scaled backbone approach. A short introduction about backbone methodology is already given in this publication and more information is provided in Ref. [II–18]. The implementation of the backbone scaling methodology as part of the GMC model is given below.

The methodology starts with the compilation of candidate GMMs (from a large proponent database) that has to satisfy a set of pre-defined qualitative rules such as those given in Ref. [II–11]. The GMC team can also set additional rules specific to the project. For the example case study, the GMC lead uses the Ref. [II–11] qualitative criteria to identify candidate GMMs. The candidate GMMs (Table II–4) are developed for shallow active crustal regions based on ground motion data in Europe, the Middle East, California, and Japan. One of the major justifications of the GMC lead and the GMC team in selecting these models is that they are developed from well-established (reliable) ground motion databases that reflect the seismo-tectonic features in the region of interest.

TABLE II–4. CANDIDATE GMMS USED IN THE DEVELOPMENT OF MEDIAN GMM VIA SCALED BACKBONE METHODOLOGY

GMM	Abbreviation	Main Region	Aleatory Variability Model
Akkar et al. (2014)	ASB14	Europe and Middle East	Constant
Bindi et al. (2014)	Bnd14	Europe and Middle East	Constant
Abrahamson et al. (2014)	ASK14	Global	M-dependent
Boore et al. (2014)	BSSA14	Global	M-dependent
Campbell and Bozorgnia (2014)	CB14	Global	M-dependent
Chiou and Youngs (2014)	CY14	Global	M-dependent
Kale et al. (2015)	KAAH15	Turkey	M-dependent
Cauzzi et al. (2015)	CFVB15	Global	Constant
Lanzano et al. (2019)	Lzn19	Italy	Constant
Kotha et al. (2020)	Kth20	Europe and Middle East	Constant
Boore et al. (2021)	Boo21	Greece	M-dependent

The CY14 model is selected as the backbone ground motion model (GMM_{BB}) among this set because its functional form is capable of modelling important ground motion features such as the hanging wall effect, nonlinear site response, basin effect, etc. Another reason to select CY14 as GMM_{BB} is that its ground motion prediction performance is verified by many studies for different shallow active seismicity earthquake scenarios [II–19, II–16].

The steps given below will be repeated for each spectral period of interest and are considered while developing the median ground motion model using the scaled backbone methodology:

(1) Obtain the spectral amplitude (SA) predictions of each candidate model for a pre-defined earthquake scenario set (magnitude–distance pairs) that are based on deaggregation results from preliminary PSHA.

(2) Calculate the logarithmic mean [$\ln\left(\overline{SA|GMM}\right)$] and logarithmic standard deviation [$\sigma_{\ln\,(SA|GMM)}$] of the SA predictions obtained from the predefined earthquake scenario set.

(3) Compute scaling factors [$\ln(SF)$] to centre the GMM_{BB} (CY14 for the example PSHA) as given in Eqn. (II–1)

$$\ln(SF) = \ln\left(\overline{SA|GMM}\right) - \ln\,(SA|GMM_{BB}) \qquad \text{(II–1)}$$

(4) Regress on $\ln(SF)$ values to develop scaling factor functional forms. The functional form principally has to consider magnitude and distance as the predictor variables because most of the time these two parameters affect the variation in $\ln(SF)$.

(5) Scale the centred GMM_{BB} up and down to model the epistemic uncertainty in median ground motion as a fraction of $\sigma_{\ln\,(SA|GMM)}$. ε, which is referred to as the number of standard deviations that will be used while defining the fractions of $\sigma_{\ln\,(SA|GMM)}$. The scaling addresses different percentiles of median SA distribution arising from the epistemic uncertainty. The median SA distribution is a good representative of CBR in median SA and different percentiles of this distribution would discretize it for its applicability to GMC logic tree framework.

For the example PSHA, the logarithmic scale factors [ln(SF)] are calculated for a set of spectral periods including PGA encompassing the periods T ≤ 1 s (T = 0.05, 0.1, 0.2, 0.3, 0.5, 0.75, 1.0 s). The GMC lead and the GMC team use discrete M-R_{RUP} (rupture distance; distance metric used in the ground motion model) pairs ranging between $5.0 \leq M_w \leq 8.0$ and $R_{RUP} \leq 50$ km that are determined from the deaggregation analyses of above discrete periods at AFEs = 10^{-4}, 10^{-5}, and 10^{-6}.

After computing the scaling factors [ln(SF)] from Eqn. (II–1), the GMC team develops the scaling functions by regressing on the scaling factors computed at each spectral discrete period. The functional form given in Eqn. (II–2) is used to develop the scaling function as ln(SF) values are dependents of magnitude and rupture distance for the discrete periods of concern. The regression coefficients ($c_0 - c_5$) of Eqn. (II–2) for each period are given in Table II–5.

$$\ln(SF) = c_0 + c_1 M_w + c_2 M_w^2 + [c_3 + c_4 \ln(R_{RUP}) + c_5 M_w] \ln(R_{RUP}) \quad \text{(II–2)}$$

TABLE II–5. REGRESSION COEFFICIENTS TO SCALE BACKBONE GMM WITH RESPECT TO EQUATION (II–2)

T (s)	c_0	c_1	c_2	c_3	c_4	c_5	$\overline{\sigma_B}$
PGA	-0.74138	0.17750	-0.01282	0.08703	-0.00714	-0.00663	0.17
0.05	-0.07904	-0.01223	-0.00046	0.04645	-0.00807	-0.00296	0.17
0.1	-0.49697	0.09709	-0.00861	0.14943	-0.00824	-0.01593	0.17
0.2	-2.25955	0.54849	-0.03609	0.22760	-0.00642	-0.02606	0.17
0.3	-2.17021	0.48838	-0.02886	0.20763	-0.00322	-0.02303	0.17
0.5	-1.43817	0.25217	-0.01122	0.15380	0.00241	-0.01431	0.17
0.75	-0.57236	-0.01519	0.00884	0.11868	0.00569	-0.00886	0.17
1.0	-0.08606	-0.17910	0.02283	0.13382	0.00541	-0.01035	0.17

As stated in the above methodology, $\sigma_{\ln(SA|GMM)}$ is used to describe the epistemic uncertainty range in median SA. This range (distribution) is covered by scaling the GMM_{BB} up and down at different percentiles identified by a set of discrete ε values. Figure. II–7 shows the period-dependent variation of $\sigma_{\ln(SA|GMM)}$ for the example PSHA as well as its period independent version $\overline{\sigma_B}$ shown by red solid line. $\overline{\sigma_B}$ takes a value of 0.17 (Table II–5) and it is in close agreement with the $\sigma_{\ln(SA|GMM)}$ variation. Therefore, the GMC lead and the GMC team decide to use $\overline{\sigma_B}$ instead of $\sigma_{\ln(SA|GMM)}$ as the differences between $\overline{\sigma_B}$ and $\sigma_{\ln(SA|GMM)}$ (i.e. the actual period-dependent logarithmic standard deviations) do not affect the precision in describing the epistemic uncertainty range of median SA. This decision is taken after some case studies performed by $\overline{\sigma_B}$ and $\sigma_{\ln(SA|GMM)}$.

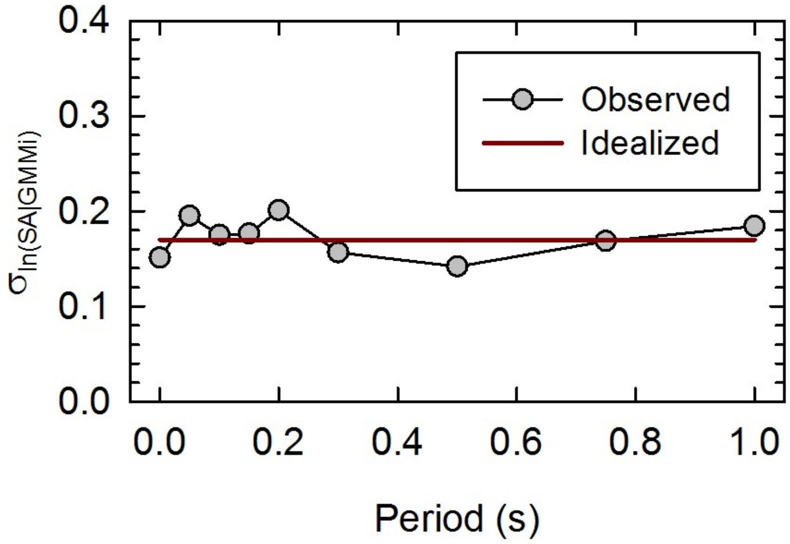


FIG. II–7. Period dependent variation and idealized form of logarithmic standard deviations.

The magnitude and distance-dependent CBR distribution of median SA is discretized by employing the 3-point discretization rule in Ref. [II–20] to develop the scaling factors for scaling GMM_{BB} (CY14 for the example PSHA). This decision by the GMC lead and the GMC team assumes a normal distribution for idealizing the median SA distribution. This rule idealizes the normal distribution at 8.5th, 50th, and 91.5th percentiles associated with the weights of 0.25, 0.50, and 0.25, respectively. The scaled backbone models are obtained using Eqn. (II–3) in which ε is the multiplier of $\overline{\sigma_B}$ to represent the 8.5th, 50th, and 91.5th percentiles. To this end, ε takes the values of -1.375, 0, and +1.375, respectively for representing the 8.5th, 50th, and 91.5th percentiles. *M* in Eqn. (II–3) takes the values 1, 2, and 3 (corresponding to 8.5th, 50th, and 91.5th percentiles, respectively) to represent the lower range, centre, and upper range of the median SA distribution. Ln(SA | GMM_{BB}) is the median SA estimate of the GMM_{BB} (CY14) while ln(SF) is determined from Eqn. (II–2).

$$\mathrm{Ln}\big(\mathrm{SA}|GMM_{B,m}\big) = \ln(\mathrm{SA}|\mathrm{GMM}_{BB}) + \ln(\mathrm{SF}) \pm \varepsilon\overline{\sigma_B} \tag{II–3}$$

Figures II–8 and II–9 show the distance-dependent variation of the scaled backbone GMM for median PGA and SA at T = 1.0 s, respectively after implementing the scaled backbone methodology. The red curves in each panel are the centred GMM_{BB} (i.e., GMM_{BB} scaled to represent 50th percentile) whereas the upper and lower black dashed lines are the scaled GMM_{BB} for the 8.5th and 91.5th percentiles. The thin grey lines show the median estimations of candidate GMMs used to develop the scaled GMMBB. The plots indicate that the range of the scaled backbone models adequately captures the range of epistemic uncertainty dictated by the candidate GMMs. Thus, the epistemic the uncertainty in median SA is modelled effectively by the scaled backbone GMM set.

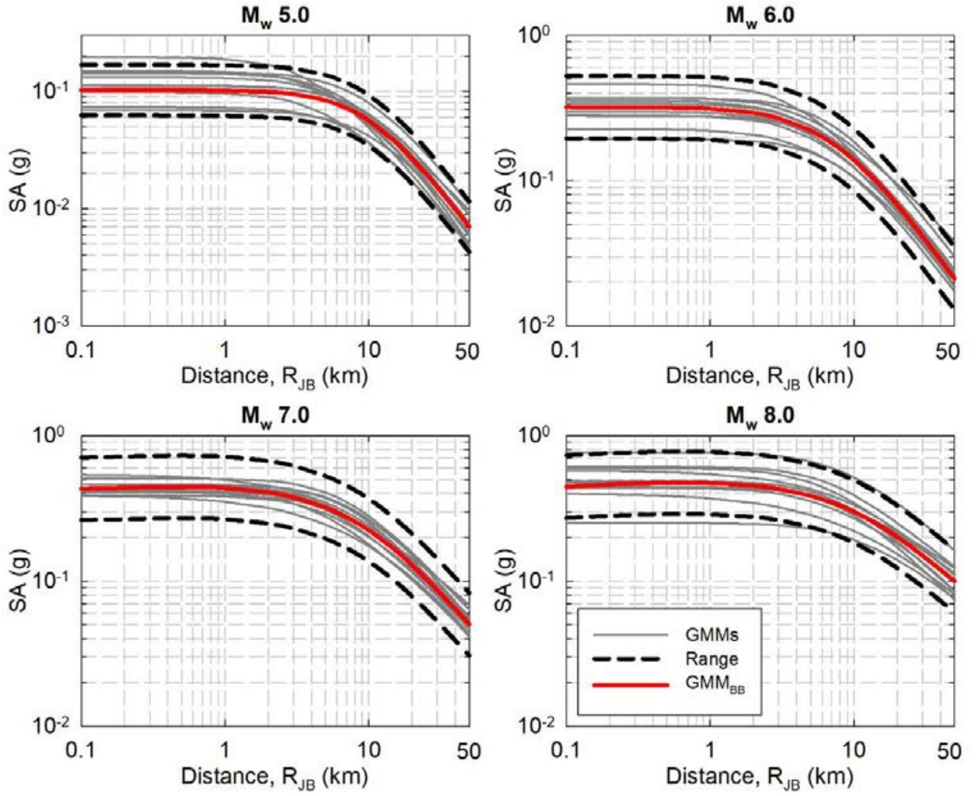


FIG. II–8. Distance-dependent variations of scaled GMMBB for median PGA at different magnitudes.

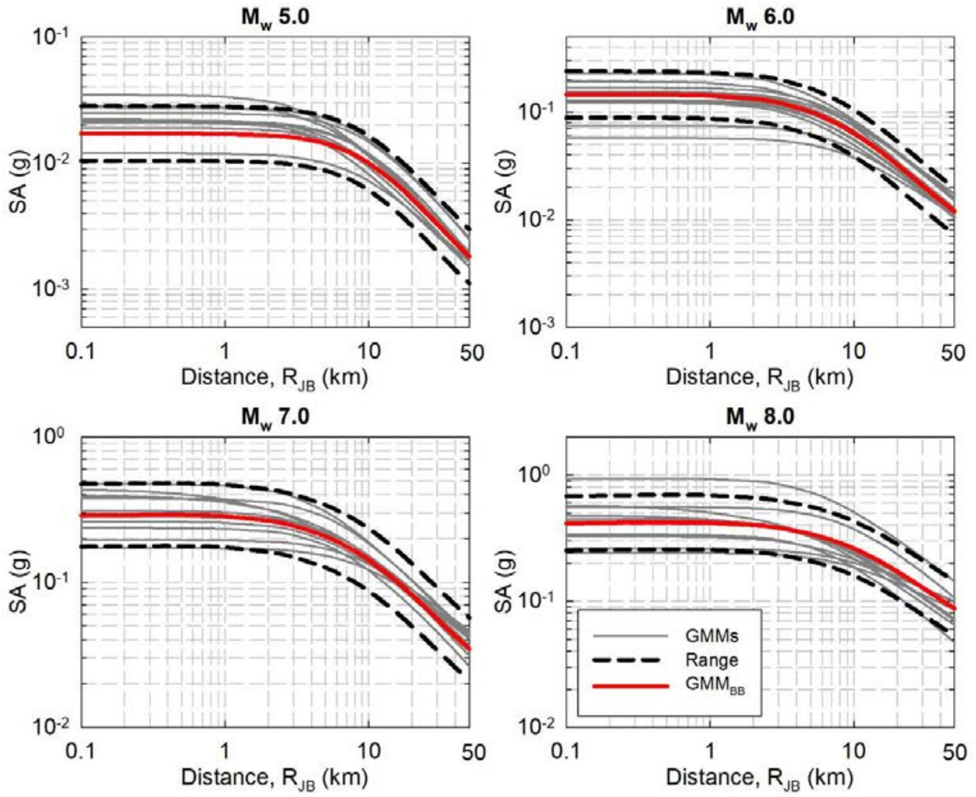


FIG. II–9. Distance-dependent variations of scaled GMMBB for median SA(T = 1.0s) at different magnitudes.

The inherent aleatory variability in the spectral amplitude estimation is described by a standard deviation (generally called "sigma"). After developing the ground motion model for median SA estimations via the scaled backbone approach, the GMC team develops the sigma model to complete the entire GMC modelling. The GMC team follows a methodology similar to the one used in median ground motion modelling while developing the sigma model.

The observations from the deaggregation analyses of preliminary PSHA suggest that large magnitude events dominate the hazard at the SMR site. Therefore, the GMC lead and the team develop a sigma model capable of accounting for the CBR in sigma at large magnitudes. The CBR in sigma needs to address the epistemic uncertainty arising from the sigma models proposed for shallow active crustal regions. The GMC lead and the team use the sigma models of candidate GMMs compiled for this project (Table II–4) to develop the sigma model, as these GMMs are decided to be good representatives of aleatory variability for spectral amplitude prediction at the SMR site. Some of the sigma models among the compiled candidate GMMs are magnitude dependent (heteroscedastic sigma models; ASK14, BSSA14, CB14, CY14, KAAH15, Boo21) whereas the other candidate GMMs provide homoscedastic sigma models disregarding the magnitude dependency in aleatory variability. The GMC lead and the team consider this specific point while developing the sigma model.

The standard deviation is composed of between-event (τ) and within-event (ϕ) standard deviation components. The period-dependent τ and ϕ models of candidate GMMs are evaluated by the GMC team for two discrete magnitudes: M_w 5.0 and M_w 6.5. The team evaluates the τ and ϕ models of all candidate GMMs at M_w 5.0 whereas only magnitude-dependent sigma models are considered for Mw 6.5. The reason of using M_w 6.5 for evaluating the τ and ϕ behaviour is that the heteroscedastic models compute smaller τ and ϕ values for $M_w \geq 6.5$ imposing a smaller aleatory variability at large magnitude events.

The grey dots in Fig. II–10 show the period-dependent variations of τ and ϕ values of candidate GMMs (upper and lower rows, respectively) for $M_w = 5$ and $M_w = 6.5$ (left and right panels, respectively). The τ and ϕ range (distribution) depicted by the candidate GMMs are considered to represent the epistemic uncertainty (CBR) in the sigma model. The red dashed lines are the median τ and ϕ values ($\tau_{B,Central}$ and $\phi_{B,Central}$, respectively) representing the 50th percentile of the τ and ϕ distributions. The dashed black lines show the 8.5th and 91.5th percentiles of the τ and ϕ distributions (lower and upper black dashed lines, respectively). They are computed by $\tau_{B,Central} \pm 1.375\tau_\sigma$ and $\phi_{B,Central} \pm 1.375\phi_\sigma$ where τ_σ and ϕ_σ are the standard deviations of the τ and ϕ distributions, respectively. The GMC lead and the team decide to use, respectively, 0.06 and 0.05 for τ_σ and ϕ_σ because the period-dependent variations of τ_σ and ϕ_σ are insignificant and the above simplification does not jeopardize the uncertainty in the sigma model.

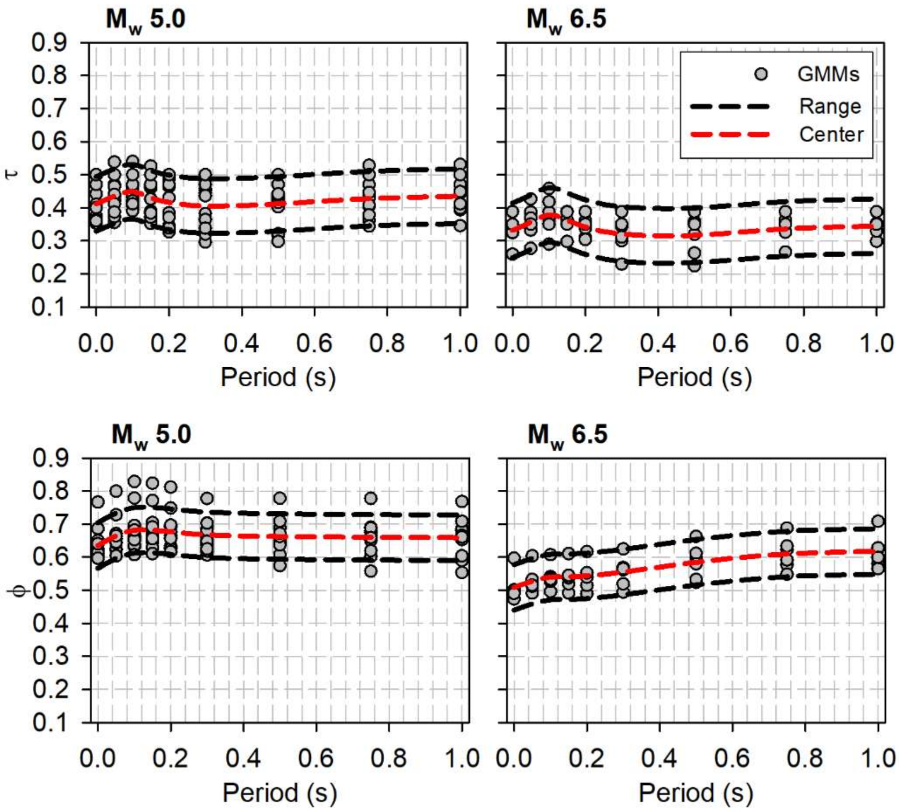


FIG. II–10. Comparison of the range of τ (top row) and ϕ (bottom row) values of the candidate GMMs.

As in the case of the median SA ground motion model, the GMC team uses the above percentiles in the 3-point discretization rule [II–20]) to idealize the CBR distribution of sigma model for its implementation in GMC logic tree. The associated weights are 0.25, 0.50, and 0.25 for the 8.5th, 50th, and 91.5th percentiles, respectively that are also given while discretizing the CBR distribution of the median SA ground motion model.

Equations (II-4) and (II-5) represent the magnitude-dependent τ and ϕ models developed by the GMC team (referred to as τ_B and ϕ_B, respectively to indicate that they are developed specific to the example PSHA). The total sigma model is the square root of the sum of the squares of τ_B and ϕ_B as given in Eqn. (II-6). σ_B is used in the PSHA integral. Table II-6 gives the period-dependent τ and ϕ values that are used in Eqns. (II-4) and (II-5).

$$\tau_B = \begin{cases} \tau_1 & M_w \leq 5.0 \\ \tau_1 + (\tau_2 - \tau_1)(M_w - 5.0)/1.5 & 5.0 < M_w < 6.5 \\ \tau_2 & M_w \geq 6.5 \end{cases} \tag{II-4}$$

$$\phi_B = \begin{cases} \phi_1 & M_w \leq 5.0 \\ \phi_1 + (\phi_2 - \phi_1)(M_w - 5.0)/1.5 & 5.0 < M_w < 6.5 \\ \phi_2 & M_w \geq 6.5 \end{cases} \tag{II-5}$$

$$\sigma_B = \sqrt{\tau_B + \phi_B} \tag{II-6}$$

TABLE II–6. PROJECT-SPECIFIC SIGMA MODEL VALUES REPRESENTING 8.5TH, 50TH, AND 91.5THPERCENTILES OF THE CBR DISTRIBUTION IN SIGMA

a) 8.5th percentile				
T(s)	τ_1	τ_2	ϕ_1	ϕ_2
0	0.3294	0.2493	0.5666	0.4399
0.05	0.3532	0.2742	0.5963	0.4588
0.1	0.3661	0.2958	0.6128	0.4710
0.2	0.3338	0.2589	0.6082	0.4751
0.3	0.3227	0.2377	0.5987	0.4865
0.5	0.3293	0.2344	0.5936	0.5158
0.75	0.3451	0.2536	0.5915	0.5410
1	0.3524	0.2619	0.5901	0.5487

b) 50th percentile				
T(s)	τ_1	τ_2	ϕ_1	ϕ_2
0	0.4119	0.3318	0.6353	0.5087
0.05	0.4357	0.3567	0.6650	0.5275
0.1	0.4486	0.3783	0.6815	0.5397
0.2	0.4163	0.3414	0.6770	0.5439
0.3	0.4052	0.3202	0.6674	0.5552
0.5	0.4118	0.3169	0.6623	0.5845
0.75	0.4276	0.3361	0.6602	0.6097
1	0.4349	0.3444	0.6589	0.6174

c) 91.5th percentile				
T(s)	τ_1	τ_2	ϕ_1	ϕ_2
0	0.4944	0.4143	0.7041	0.5774
0.05	0.5182	0.4392	0.7338	0.5963
0.1	0.5311	0.4608	0.7503	0.6085
0.2	0.4988	0.4239	0.7457	0.6126
0.3	0.4877	0.4027	0.7362	0.6240
0.5	0.4943	0.3994	0.7311	0.6533
0.75	0.5101	0.4186	0.7290	0.6785
1	0.5174	0.4269	0.7276	0.6862

FIG. II–11 shows the overall GMC logic-tree framework proposed by the GMC lead and the team after performing the above calculations for median SA and sigma ground motion models. Essentially, the methodology implemented results in nine distinct ground motion models to cover the epistemic uncertainty range (CBR) of the spectral amplitude predictions at the SMR site for T ≤ 1.0 s, including PGA.

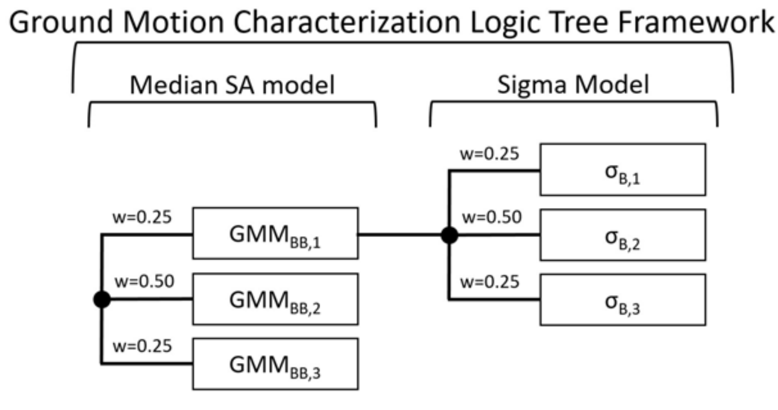


FIG. II–11. Representation of GMC logic-tree framework.

The duration of ground motion modelling as explained above is anticipated as three to four months if the GMC lead holds regular and intensive meetings with the GMC team members as well as the technical integrator. As indicated in the technical team profiles the GMC team members need to be aware of the details of ground motion modelling to prevent delays in the project duration.

II–8.3. Seismic Source Characterization Model

The seismic source characterization model is established for the seismic sources confined within a radius of R = 50 km based on the deaggregation analyses of the preliminary PSHA. The seismic source characterization lead and the seismic source characterization team consider the following items to capture the CBR in the seismic source characterization model:

(1) Consideration of alternative source models to capture the uncertainties related to source geometry and rupture characteristics;
(2) Consideration of alternative probability distributions for hypocentral depth locations;
(3) Consideration of a set of dip angles for areal and fault sources on the basis of seismic data and literature survey;
(4) Consideration of different magnitude-rupture area scaling relationships;
(5) Uncertainty in M_{max};
(6) Consideration of alternative magnitude recurrence models for fault sources;
(7) Consideration of different slip rates for fault sources based on the literature survey.

The seismic source characterization model consists of seven alternative areal source and six alternative fault source models. The seismic source characterization team develops these models from a detailed literature survey about the seismicity of the region of interest as well as its seismotectonic structure. The geometries of the original areal sources compiled from this literature survey are modified for R = 50 km. Consequently, the original activity rates of these areal sources are also adjusted by considering the modified geometries. The seismic source characterization team implements a similar approach for the background seismicity component of the fault source modelling. Figure. II–12 shows an example of modifying the original areal sources (right panel) for R = 50 km, the radial distance used to limit the site specific PSHA model. The modified areal sources are presented on the left panel. The activity rates of the original areal sources are adjusted for the modified areas of each areal source.

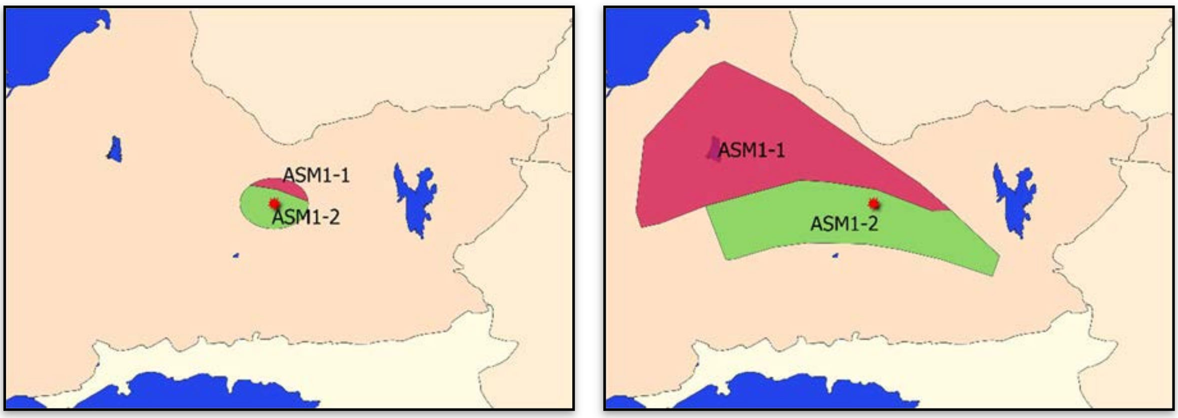


FIG. II–12. Areal Source Model 1 (ASM1) (left panel) and its original geometry (right panel).

The alternative areal source models consist of either two areal sources (as depicted in Fig. II–12) or three areal sources. Figure II–13 shows the generic geometry representation of the latter case. These alternative areal source models account for the differences in seismic activity for the region of interest. The areal source models labelled as ASM1, ASM2, ASM4 and ASM6 consist of two areal sources whereas the rest of the areal models consider three areal sources. The areal source zones in each areal source model are labelled with numbers according to their positions. For example, the source zones of ASM1 given in Fig. II–12 are designated as ASM1-1 and ASM1-2 starting numbering from the northernmost source. In as similar manner, the source zones in ASM3 are ASM3-1, ASM3-2 and ASM3-3 from top to bottom. The first area source zone (ASM1-1, ASM2-1, ASM3-1,…,ASM7-1) is identical in all source models although its name changes between source models.

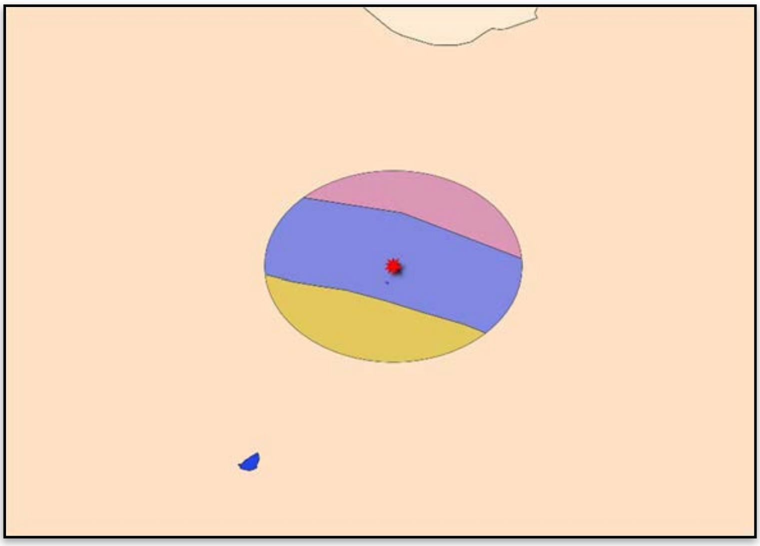


FIG. II–13. Areal source model geometry that is common for ASM3, ASM6 and ASM7.

FIG. II–14 shows a typical area source model logic tree framework developed by the seismic source characterization lead and the team. The logic tree framework is complicated and covers the CBR for this area source model considering the entire information gathered from various sources including literature, local experts and seismicity in the region. In other words, the preliminary PSHA and the consequent deaggregation analyses are employed to decide on the geographic boundaries of the study area for seismic sources that control the vibratory ground motion hazard at the site of interest. The seismic source characterization lead and the seismic source characterization team use this information to address the full CBR in source modelling uncertainty. The merit of graded approach is to focus on the most critical sources in seismic source characterization modelling.

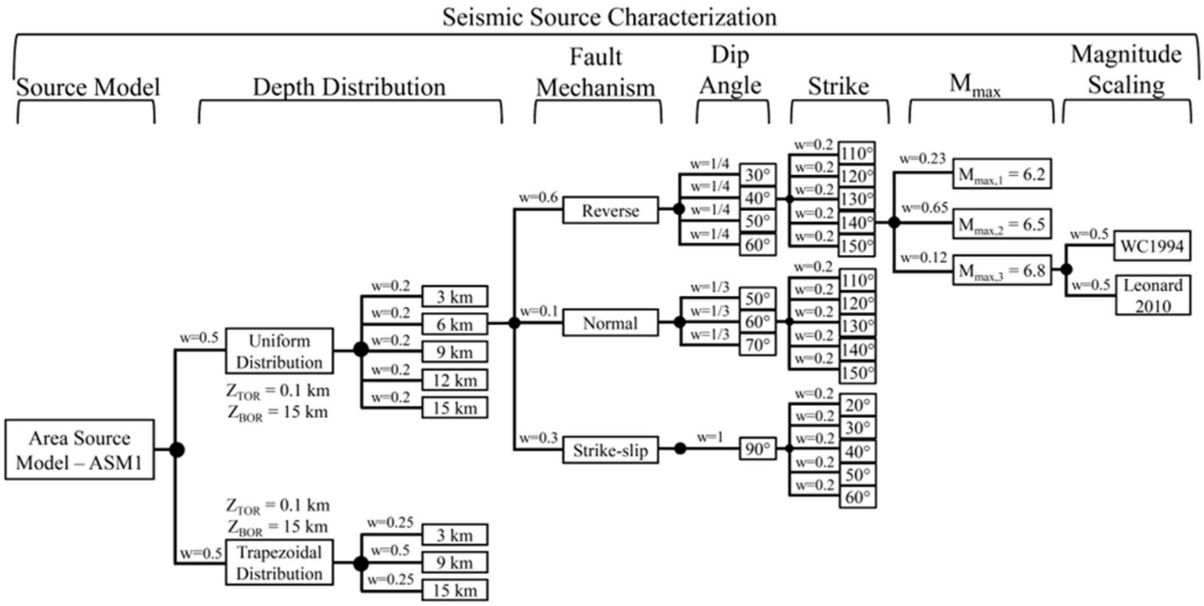


FIG. II–14. An example seismic source characterization logic-tree framework for an area source model (ASM1).

The duration of seismic source characterization modelling is anticipated to be four to five months provided that the preliminary PSHA reaches its objectives (delineation of controlling seismic sources for final seismic source characterization modelling). The regular and intensive interactions between the seismic source characterization team members as well as the technical integrator is another key aspect to prevent possible delays in the anticipated duration.

The fault source models in the seismic source characterization model provide different alternatives in terms of geometry and fault source parameters. The FSM1, FSM2 and FSM3 models consider the segmental structure of the faults whereas the remaining three models disregard some of the segmentation in FSM1, FSM2 or FSM3 to account for the possibility of progressive ruptures. Fig. II–15 shows an example case for segmented and unsegmented models. The segmented and unsegmented fault models also differ among themselves by fault characteristics such as dipping directions. The segmented type fault models incorporate 15 faults whereas the unsegmented models comprise of 10 faults. The fault source parameters considered in the segmented models are given in Table II–7. As depicted in this Table, the difference between the segmented models only shows itself in the FS1 fault whose style-of-faulting and dipping directions change among the alternative models (FSM1, FSM2 and FSM3). The other fault characteristics are same in all segmented models. The fault model names corresponding to each alternative of FS1 are indicated under the "Style of Faulting" column in Table II–7.

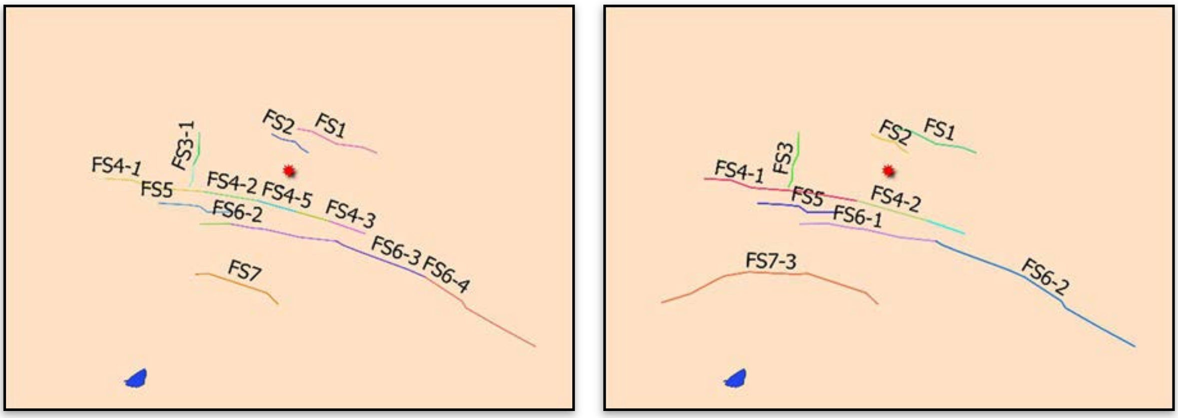


FIG. II–15. The segmented (left panel) and unsegmented fault source models (right panel).

TABLE II–7. SOURCE CHARACTERISTICS OF SEGMENTED TYPE FAULT SOURCE MODELS

Fault Name	Seismogenic Thickness (km)	Fault Length (km)	$Mmax_{min}$, $Mmax_{mean}$, $Mmax_{max}$	M_{char}	Style of Faulting	Dip Angles (degree)	Slip Rate (mm/yr)
FS1	0.1-25	31	6.2, 6.5, 6.8	6.5	R-South Dipping (FSM1)	40, 50, 60, 70	0.068, 0.08
					R-North Dipping (FSM2)	55, 65, 75, 85	
					SS (FSM3)	70, 80, 90	
FS2	0.1-20	17	5.8, 6.1, 6.4	6.1	N	40, 50, 60	0.042, 0.05
FS3-1	0.1-20	10	5.5, 5.8, 6.1	5.8	N	50, 60, 70	0.074, 0.08
FS3-2	0.1-20	17	5.8, 6.1, 6.4	6.1	N	50, 60, 70	0.074, 0.08
FS4-1	0.1-20	36	6.3, 6.6, 6.9	6.6	R	30, 40, 50, 60	0.1, 0.2
FS4-2	0.1-20	21	6, 6.3, 6.6	6.3	R	30, 40, 50, 60	0.1, 0.2
FS4-3	0.1-20	16	5.9, 6.2, 6.5	6.2	R	30, 40, 50, 60	0.1, 0.2
FS4-4	0.1-20	11	5.6, 6.2, 6.5	5.9	R	30, 40, 50, 60	0.1, 0.2
FS4-5	0.1-20	16	5.9, 6.2, 6.5	6.2	R	30, 40, 50, 60	0.1, 0.2
FS5	0.1-20	33	6.2, 6.5, 6.8	6.5	R	30, 40, 50, 60	0.12, 0.25
FS6-1	0.1-20	14	5.8, 6.1, 6.4	6.1	R	30, 40, 50, 60	0.28, 0.4
FS6-2	0.1-20	38	6.3, 6.6, 6.9	6.6	R	30, 40, 50, 60	0.28, 0.4
FS6-3	0.1-20	36	6.3, 6.6, 6.9	6.6	R	30, 40, 50, 60	0.28, 0.4
FS6-4	0.1-20	54	6.5, 6.8, 7.1	6.8	R	30, 40, 50, 60	0.28, 0.4
FS7	0.1-20	35	6.3, 6.6, 6.9	6.6	R	30, 40, 50	0.26, 0.36

The fault characteristics of the unsegmented models are presented in Table II–8. As in the case of segmented models, the FS1 fault is the only fault featuring different dipping directions and styles of faulting among the alternative unsegmented models. To this end, the unsegmented model names are labelled by considering the different FS1 characteristics as given in the "Style of Faulting" column in Table II–8. The characteristics of other faults are the same in all unsegmented models.

TABLE II–8. SOURCE CHARACTERISTICS OF UNSEGMENTED TYPE FAULT SOURCE MODELS

Fault Name	Seismogenic Thickness (km)	Fault Length (km)	$Mmax_{min}$, $Mmax_{mean}$, $Mmax_{max}$	M_{char}	Style of Faulting	Dip Angles (degree)	Slip Rate (mm/yr)
FS1	0.1-25	31	6.2, 6.5, 6.8	6.5	R-South Dipping (FSM4)	40, 50, 60, 70	0.068, 0.08
					R-North Dipping (FSM5)	55, 65, 75, 85	
					SS (FSM6)	70, 80, 90	
FS2	0.1-20	17	5.8, 6.1, 6.4	6.1	N	40, 50, 60	0.042, 0.05
FS3	0.1-20	27	6.1, 6.4, 6.7	6.8	N	50, 60, 70	0.074, 0.08
FS4-1	0.1-20	57	6.5, 6.8, 7.1	6.8	R	30, 40, 50, 60	0.1, 0.2
FS4-2	0.1-20	27	6.1, 6.4, 6.7	6.4	R	30, 40, 50, 60	0.1, 0.2
FS4-3	0.1-20	16	5.9, 6.2, 6.5	6.2	R	30, 40, 50, 60	0.1, 0.2
FS5	0.1-20	33	6.2, 6.5, 6.8	6.5	R	30, 40, 50, 60	0.12, 0.25
FS6-1	0.1-20	52	6.5, 6.8, 7.1	6.8	R	30, 40, 50, 60	0.28, 0.4
FS6-2	0.1-20	90	6.8, 7.1, 7.4	7.1	R	30, 40, 50, 60	0.28, 0.4
FS7-3	0.1-20	86	7.2, 7.5, 7.8	7.5	R	30, 40, 50	0.26, 0.36

The fault models and their characteristics given in Tables II–7 and II-8 reveal a complicated seismic source characterization logic tree framework and it is not given here for the sake of simplicity. All fault sources consider two alternative magnitude recurrence models associated with equal weights: the truncated Gutenberg-Richter model and the characteristic model [II–12]. Modelling uncertainty in magnitude-rupture area scaling is accounted for by using the relationships in Refs [II-21, II-22] and equal weights are given to these alternatives. The trapezoidal distribution is assigned for M_{max} values in the case of the truncated recurrence model. A single characteristic magnitude (M_{char}) value is considered for the characteristic recurrence model. The seismic source characterization team decides on the style of faulting weights relying on the literature and regional seismic data and they change from one model to the other. The alternative dip angles of the faults are equally weighted as no compelling justification exists on the dominancy of one of the alternative dip angles. The seismic source characterization team considers trapezoidal distribution for slip rates.

The complex seismic source characterization logic tree framework comprising of areal source and fault source models can be judged as detailed. However, this is a common practice in most site specific PSHA modelling of nuclear installations as the main goal is a full description of

CBR arising from the epistemic uncertainty in source modelling parameters. This practice, which is also implemented for the PSHA case study, results in a massive number of logic tree branches that complicate the PSHA modelling process and consume a significant amount of time in the analyses and post-processing of analyses. As stated, while explaining the overall structure of the graded approach, the TI, the seismic source characterization lead and the seismic source characterization team aim to cover the entire modelling uncertainty within the radial area (described by R = 50 km in this PSHA example) delineated in the deaggregation analyses of preliminary PSHA. However, as it will be discussed in the next section, further simplifications in the seismic source characterization logic tree structure are possible to increase the efficiency of the graded approach without jeopardizing the reliability as well as the accuracy in hazard results.

Figure II–16 shows the entire logic tree framework of the PSHA model. The seismic source characterization component in the illustration is compressed because each one of the areal and fault source models consists of a comprehensive logic tree structure as presented in Fig. II–14 and discussed in the above paragraphs. The following section provides the PSHA results. The significance of seismic source characterization and GMC modelling parameters, and particularly, the implications of the sophisticated seismic source characterization logic tree framework on the PSHA results are discussed in the following section.

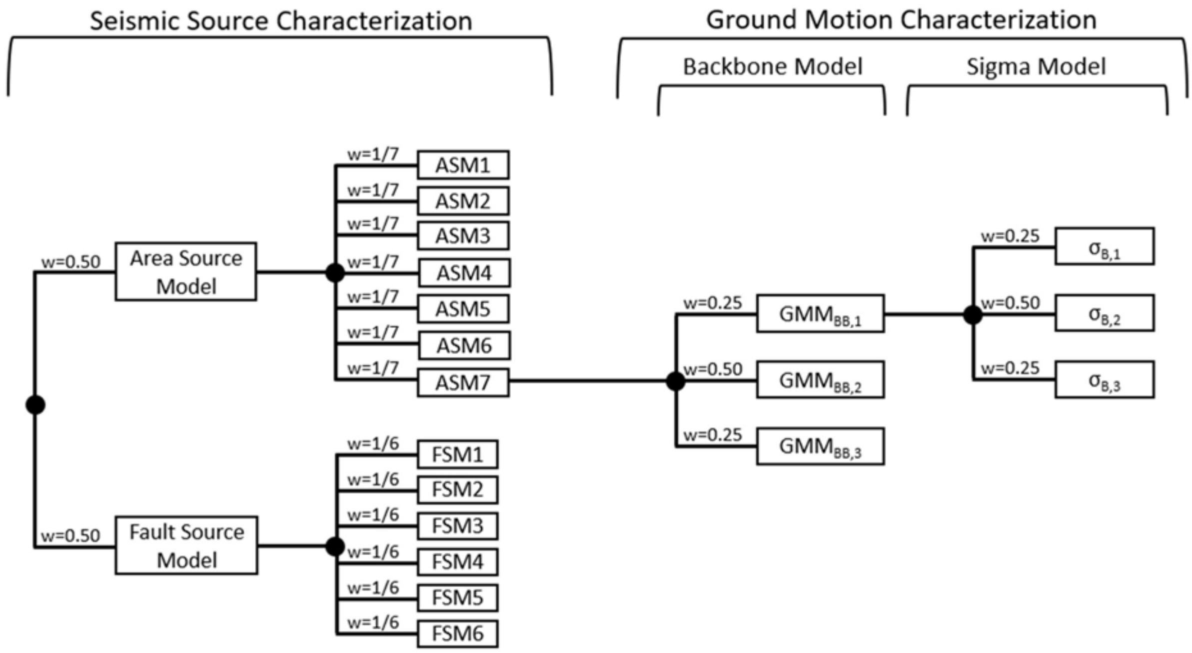


FIG. II–16. The simplified version of the combined seismic source characterization *and GMC models considered in the PSHA model.*

II–8.4. Probabilistic Seismic Hazard Analysis Results, Comparisons and Tornado Diagrams

This subsection summarizes the PSHA results and presents observations in a comparative format to particularly suggest possible simplification strategies in the seismic source characterization modelling for time-wise computational efficiency in the graded approach. The hazard analyst performs the PSHA calculations via OpenQuake software [II-23]. The first part of this section treats the areal source models and it is followed by the fault source models.

II-8.4.1. Interpretations on the probabilistic seismic hazard analysis results of areal source models

Table II–9 presents the magnitude-recurrence parameters and minimum activity rates for areal source models. It suggests that the areal source models can be grouped according to their geometries and activity rates. Accordingly, the areal source models with two source zones that are identical in terms of geometry are designated as a Type I source geometry. Similarly, the other areal source models with three source zones are labelled as Type II source models in terms of source geometry. In this connection, ASM1 and ASM2 attain very similar activity rates and their geometries are identical. Similarly, ASM3 and ASM5 have closer activity rates and they have the same source geometry. The geometry of ASM7 is the same as ASM3 and ASM5 but it represents higher seismic activity rate. On the basis of these observations, the hazard results of areal source models are investigated at some representative spectral periods. The left panels in Figures II–17 and II–18 illustrate the hazard results of all areal source models whereas the middle and the right panels in these figures show the ratio of spectral accelerations for some of the areal source model pairs classified accordingly.

Figures II–17 and II–18 reveal that the areal models with Type II source geometry (ASM3, ASM5 and ASM7) yield lower seismic hazard values compared to the models with Type I source geometry. The normalized hazard results presented in the middle and right parts of Figures II–17 and II–18 depict that a model with the same geometry and similar activity rates yield close hazard results such as ASM2-ASM1 and ASM5-ASM3 because ASM2/ASM1 and ASM5/ASM3 ratios are between 0.95-1.0 for most of the AFEs. This observation suggests that the activity rate is the only controlling parameter of the hazard for areal source models if their source zone geometries are similar and depict similar attributes in the other important source model parameters such as M_{max}. The areal source models SM3 and SM7 that have the same source zone geometry but a different activity rate result in different hazard levels strengthening the above observation. The right panels in Figures II–17 and II–18 show that the hazard ratios of SM7/SM3 are between 1.0-1.1 for PGA and 1.0-1.2 for T=1.0 s.

TABLE II–9. MAGNITUDE-RECURRENCE RELATIONSHIPS OF THE AREAL SOURCE MODELS

		a-value	b-value	Mmin	N(M>Mmin)	Total N(M>Mmin)	Source Geometry	$Mmax_{min}$-$Mmax_{max}$
All Sources	ASM1-1	1.31	0.93	5	0.000449	-	-	6.2-6.8
ASM1	ASM1-2	2.93	1.01	5	0.007973	**0.008421**	Type I	7-7.4
ASM2	ASM2-2	3.00	1.03	5	0.007225	**0.007673**	Type I	7-7.4
ASM3	ASM3-2	2.56	1.00	5	0.003838	**0.006502**	Type II	5.9-7.4
	ASM3-3	2.09	0.95	5	0.002216			7-7.4
ASM4	ASM4-2	2.96	1.05	5	0.005225	**0.005674**	Type I	7-7.4
ASM5	ASM5-2	2.62	1.01	5	0.003898	**0.006064**	Type II	5.9-7.4
	ASM5-3	2.44	1.04	5	0.001717			7-7.4
ASM6	ASM6-2	3.05	1.02	5	0.009060	**0.009508**	Type I	7-7.4
ASM7	ASM7-2	2.56	0.99	5	0.003916	**0.008133**	Type II	5.9-7.4
	ASM7-3	2.78	1.04	5	0.003769			7-7.4

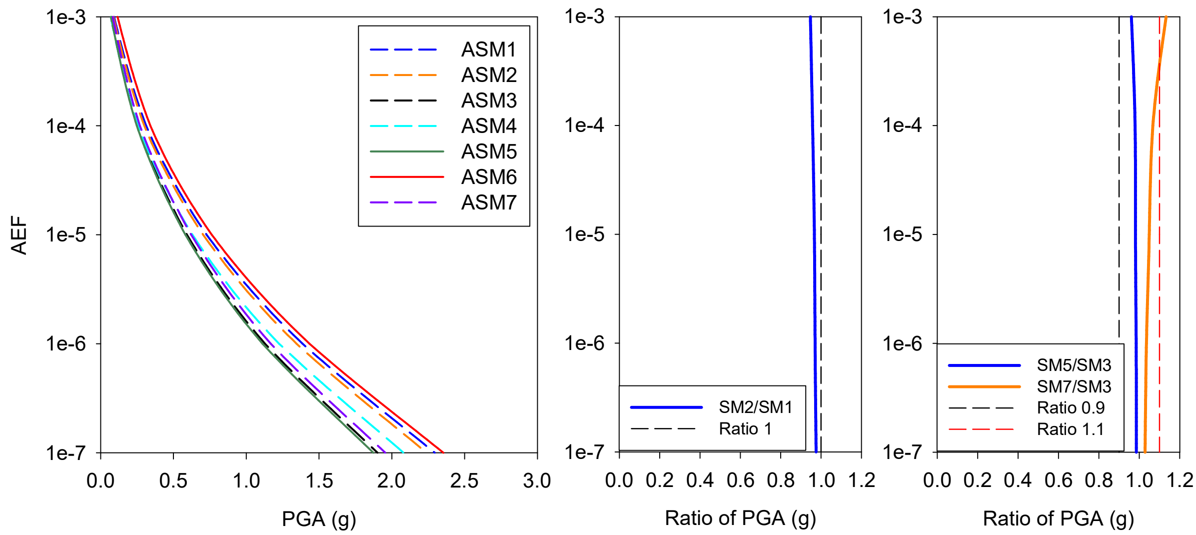


FIG. II–17. Comparison of hazard curves (left) and normalized spectral accelerations (middle and right) for the selected source models at PGA.

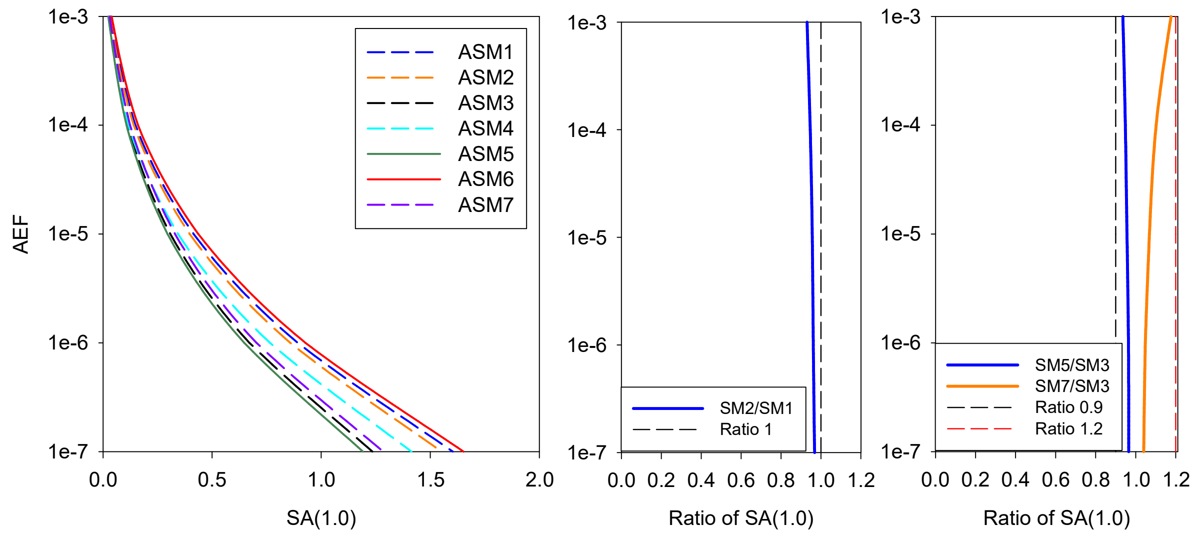


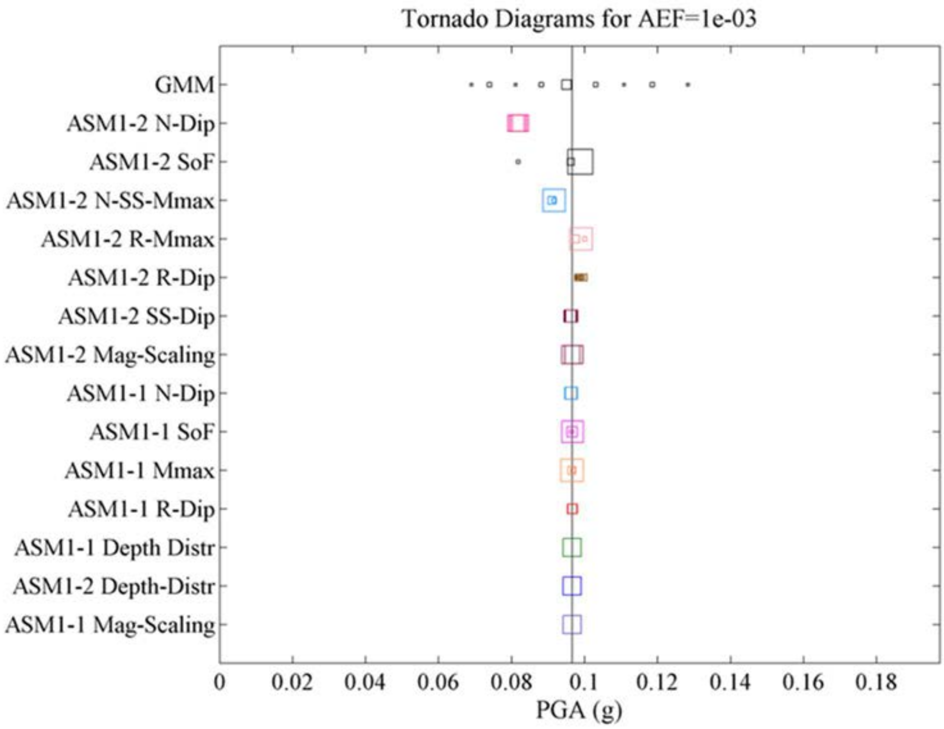
FIG. II–18.Comparison of hazard curves (left) and normalized spectral accelerations (middle and right) for the areal source models at T=1s.

The effect of each logic tree branch can be assessed through Tornado diagrams, which serve as a conventional sensitivity analysis tool in PSHA modelling to understand how the lower and upper bounds of the independent variables in the PSHA integral affect the hazard outcomes (as dependent variables) [II–23, II-24]. In other words, the Tornado diagrams evaluate the significance of seismic hazard model components (independent variables) on the mean hazard at a target AFE. They show the degree of importance of each independent variable in the logic tree by varying its upper and lower bounds while keeping the remaining variables constant, calculating the hazard, and showing the corresponding difference with respect to the mean hazard. The independent parameter having the largest difference is listed at the top of the Tornado diagram. Each square in the Tornado diagrams represents the results of the alterative branch at the considered level of the logic tree. The size of the square represents the size of the difference with respect to the mean hazard and it is adjusted according to the logic tree weights of each independent variable.

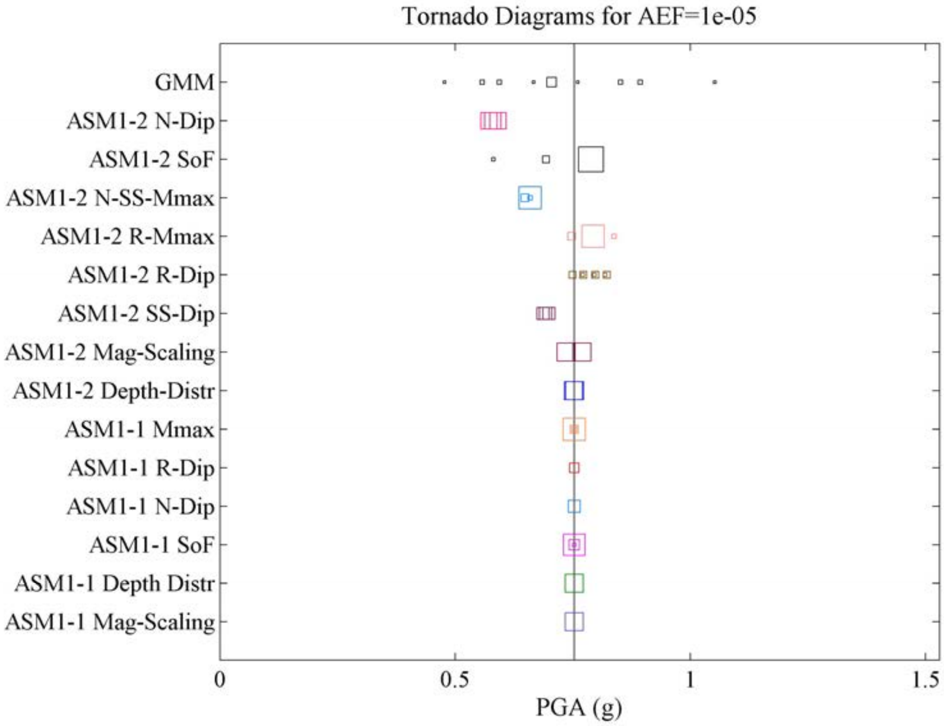
Fig. II–19 shows the Tornado plots of ASM1 for PGA at AFEs of 10^{-3}, 10^{-5} and 10^{-7} respectively. Similarly, the Tornado plots of ASM1 for T = 1s are given in Fig. II–20 for the same AFE levels. In the Tornado plots of ASM1, the investigated parameters are clarified below together with the abbreviations given y axis of each plot. Except for the GMM parameter, all investigated parameters have a prefix that defines the investigated source zone name, such as ASM1-1 and ASM1-2.

(1) GMM: The effect of Ground Motion Model, the Median SA Model and Sigma model pairs defined in GMC part of the study (9 branches);
(2) SoF: The effect of style of faulting on the source designated in the prefix;
(3) N-Dip: The effect of dip angle on normal type faulting for the source designated in the prefix;
(4) R-Dip: The effect of dip angle on reverse type faulting for the source designated in the prefix;
(5) SS-Dip: The effect of dip angle on strike-slip type faulting for the source designated in the prefix;
(6) R-M_{max}: The effect of M_{max} on reverse type faulting for the source designated in the prefix;
(7) N-SS-M_{max}: The effect of M_{max} on normal and strike-slip type faulting for the source designated in the prefix;
(8) Mag Scaling: The effect of magnitude rupture scaling law for the source designated in the prefix;
(9) Depth Distr: The effect of uniform or trapezoidal depth distribution for the source designated in the prefix.

a)



b)



c)

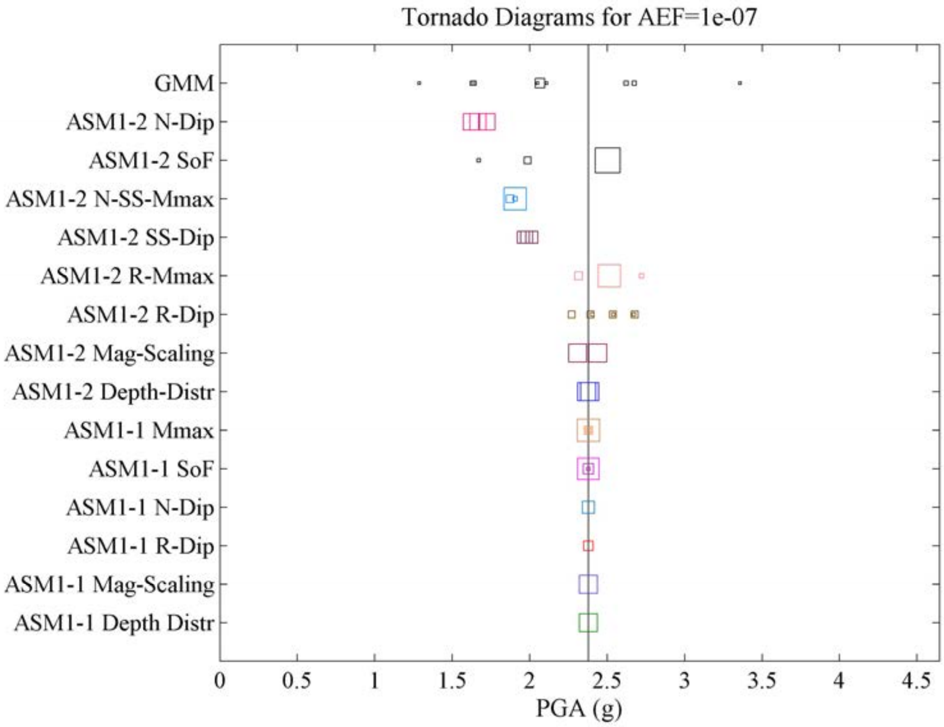
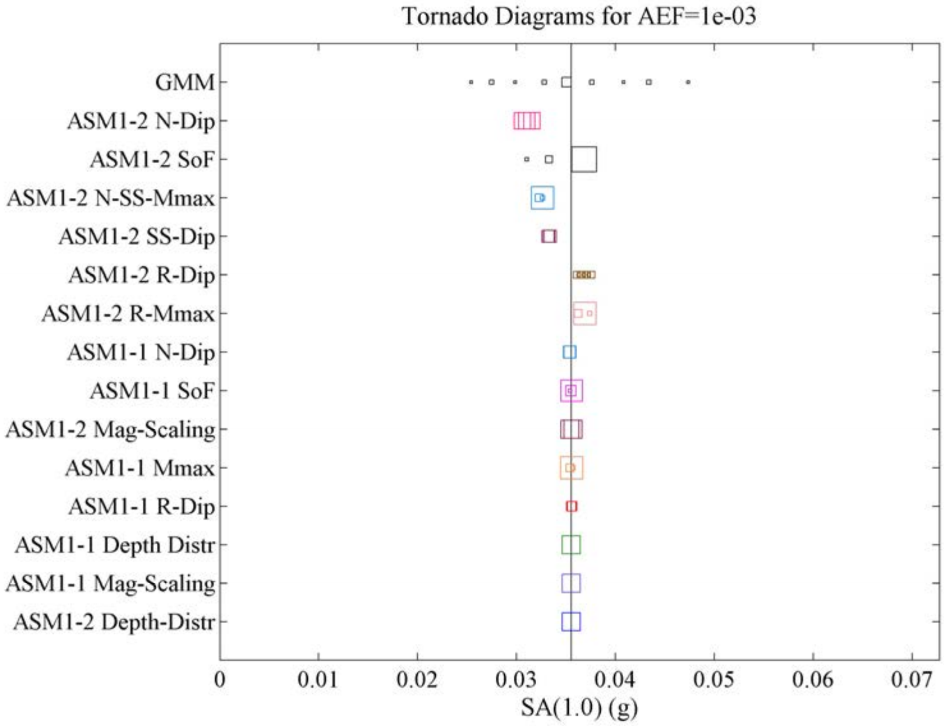
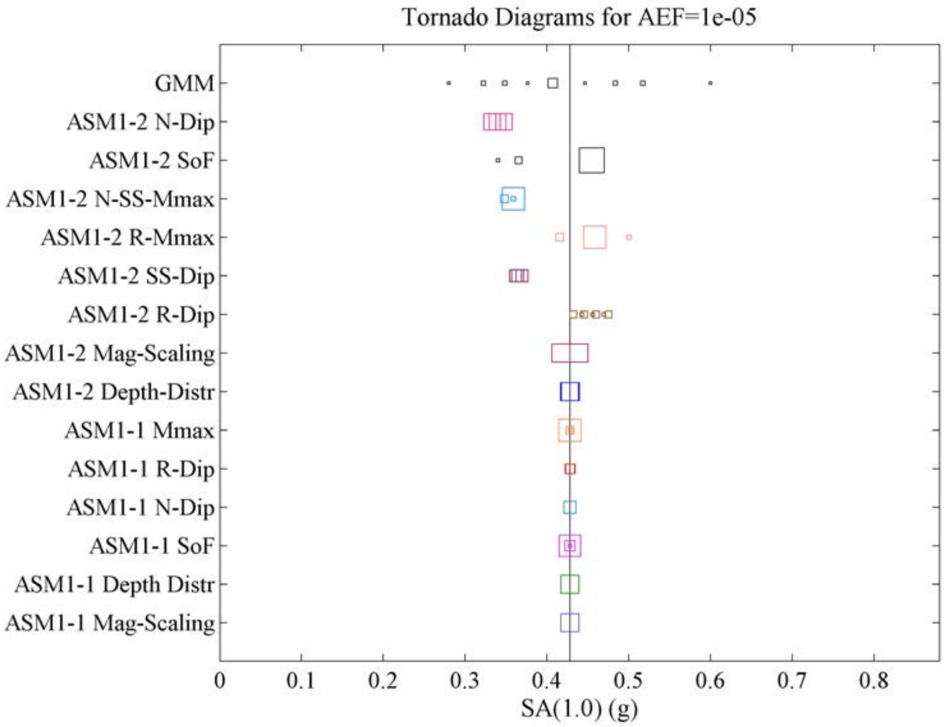


FIG. II–19. Tornado Plots of ASM1 for PGA at AFE levels of a) 10^{-3} b) 10^{-5} c) 10^{-7}.

a)


Tornado Diagrams for AEF=1e-03
GMM
ASM1-2 N-Dip
ASM1-2 SoF
ASM1-2 N-SS-Mmax
ASM1-2 SS-Dip
ASM1-2 R-Dip
ASM1-2 R-Mmax
ASM1-1 N-Dip
ASM1-1 SoF
ASM1-2 Mag-Scaling
ASM1-1 Mmax
ASM1-1 R-Dip
ASM1-1 Depth Distr
ASM1-1 Mag-Scaling
ASM1-2 Depth-Distr
0
0.01
0.02
0.03
0.04
0.05
0.06
0.07
SA(1.0) (g)

b)


Tornado Diagrams for AEF=1e-05
GMM
ASM1-2 N-Dip
ASM1-2 SoF
ASM1-2 N-SS-Mmax
ASM1-2 R-Mmax
ASM1-2 SS-Dip
ASM1-2 R-Dip
ASM1-2 Mag-Scaling
ASM1-2 Depth-Distr
ASM1-1 Mmax
ASM1-1 R-Dip
ASM1-1 N-Dip
ASM1-1 SoF
ASM1-1 Depth Distr
ASM1-1 Mag-Scaling
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
SA(1.0) (g)

c)

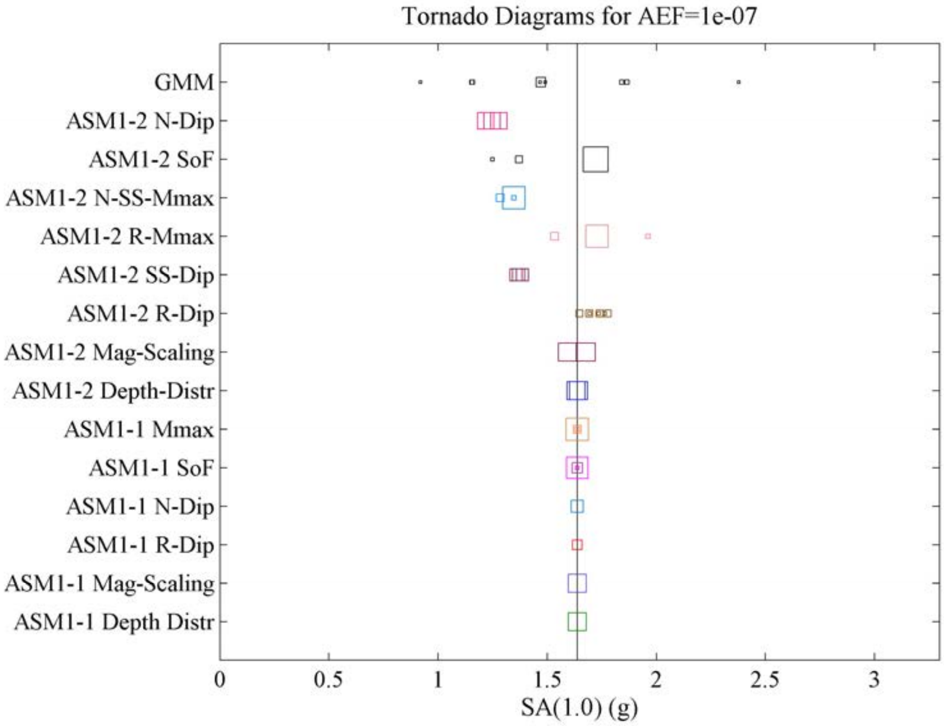


FIG. II–20. Tornado Plots of ASM1 for T = 1.0 s at AFE levels of a) 10-3 b) 10-5 c) 10-7.

The Tornado plots in Figures II–19 and II–20 indicate that the epistemic uncertainty of the source modelling parameters for ASM1-1, which has lower M_{max} value compared to its alternatives, has an insignificant impact on the hazard results. On the other hand, the same figures show that the epistemic uncertainties in the source modelling parameters of ASM1-2 have larger impact on the hazard results as this areal source model plays a role in controlling the hazard at the SMR site. Among the source modelling parameters of ASM1-2, the style-of-faulting has a greater impact on the results. For each alternative style-of-faulting in ASM1-2, the uncertainty in dip angle becomes important in the results that is more visible at the intermediate periods. Nevertheless, the impact source modelling uncertainties in ASM1-2 are still limited when compared to the impact of epistemic uncertainty of the ground motion model. As one can infer from the Tornado plots in Figures II–19 and II–20, the epistemic uncertainty in the ground motion model controls the mean hazard at all AFE levels.

II-8.4.2. Interpretations on the probabilistic seismic hazard analysis results of fault source models

The fault source models and the relative importance of epistemic uncertainties described by the seismic source characterization and GMC logic tree frameworks of fault sources are investigated in this section. As in the case of previous section, the first part presents the hazard results in terms of hazard curves and the normalized SAs of fault models. Then, the representative Tornado plots will be given for some selected fault models.

The left and middle panels in Figures II–21 and II–22 indicate that the segmented and unsegmented fault source models impact the hazard at the period range of concern in this study particularly as AFEs attain smaller values. The segmented and unsegmented models are influenced even more by the source characterization of the single fault FS1 whose style-of-faulting and dipping directions change between alternative models. These observations suggest that for fault source models, the source characterization plays an important role in the hazard results. The degree of this influence is assessed further in Tornado analyses as discussed below.

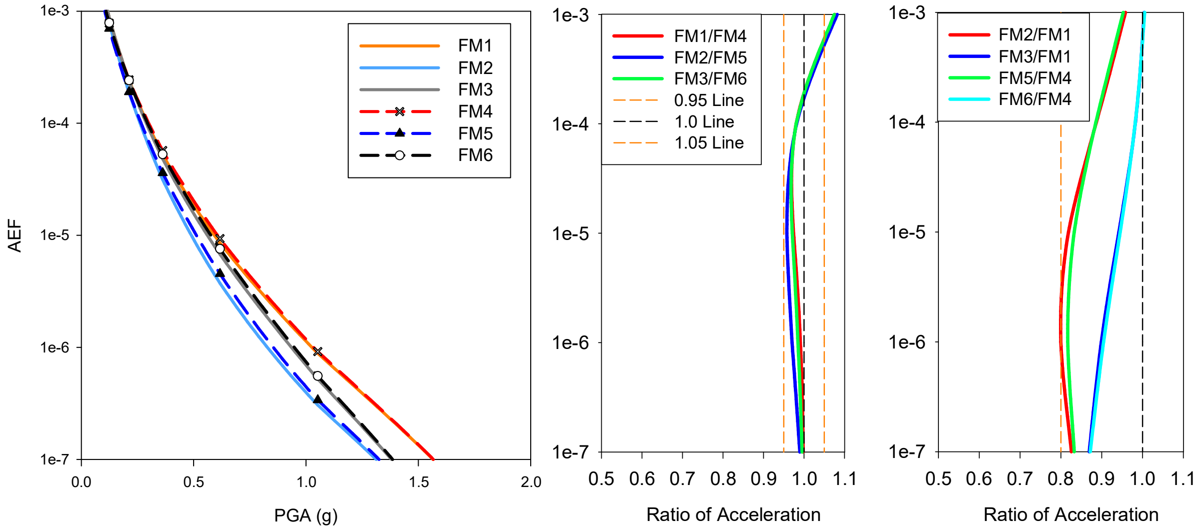


FIG. II–21. Comparison of hazard curves (left panel) and spectral acceleration ratios (middle and right panels) between alternative fault source models for PGA.

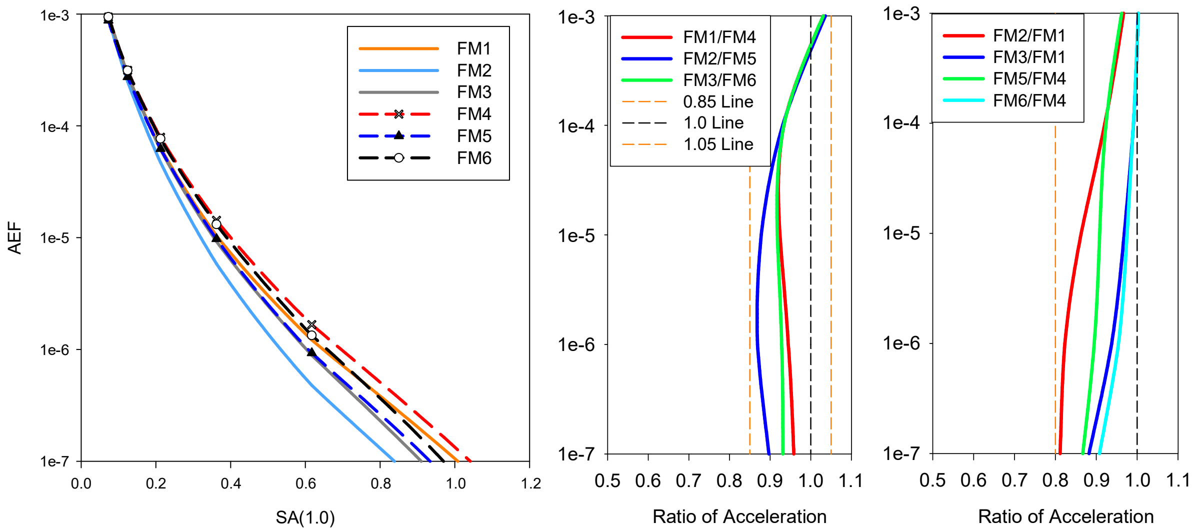


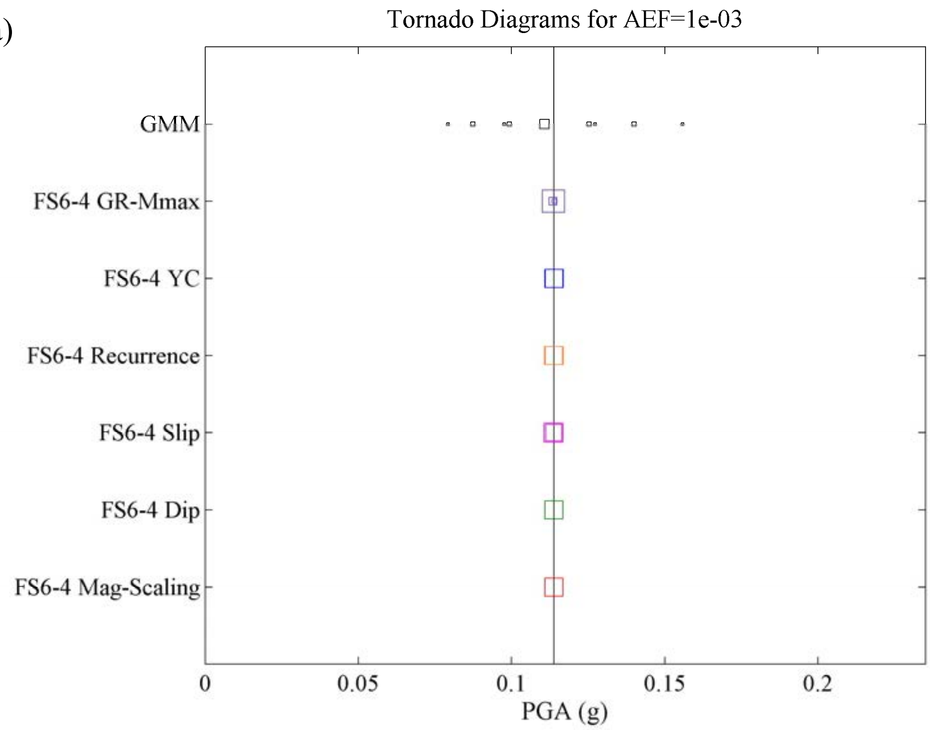
FIG. II–22. Comparison of hazard curves (left) and spectral acceleration ratios (middle and right) between alternative fault source models for T=1s.

Figures II–23 and II–24 illustrate the Tornado plots performed for the FS6-4 fault in the FSM1 model at AFEs of 10^{-3}, 10^{-5} and 10^{-7} for PGA and T = 1.0 s, respectively. The source characteristics of FS6-4 given in Table II–7 suggest that this source may have a significant contribution to the total hazard as its M_{max} range depicts larger magnitudes. However, Figs II–23 and II–24 suggest this fault source has almost no impact on the mean hazard curve, which can be explained by its remote location with respect to the SMR site: the shortest distance between FS6-4 and the SMR site is approximately 72 km. Accordingly, it can be stated that a consideration of epistemic uncertainty for remotely located sources would not improve the hazard results at the SMR site. The Tornado diagrams in these figures also indicate that the ground motion model and the epistemic uncertainty of the ground motion model are the primary components affecting the hazard at the SMR site.

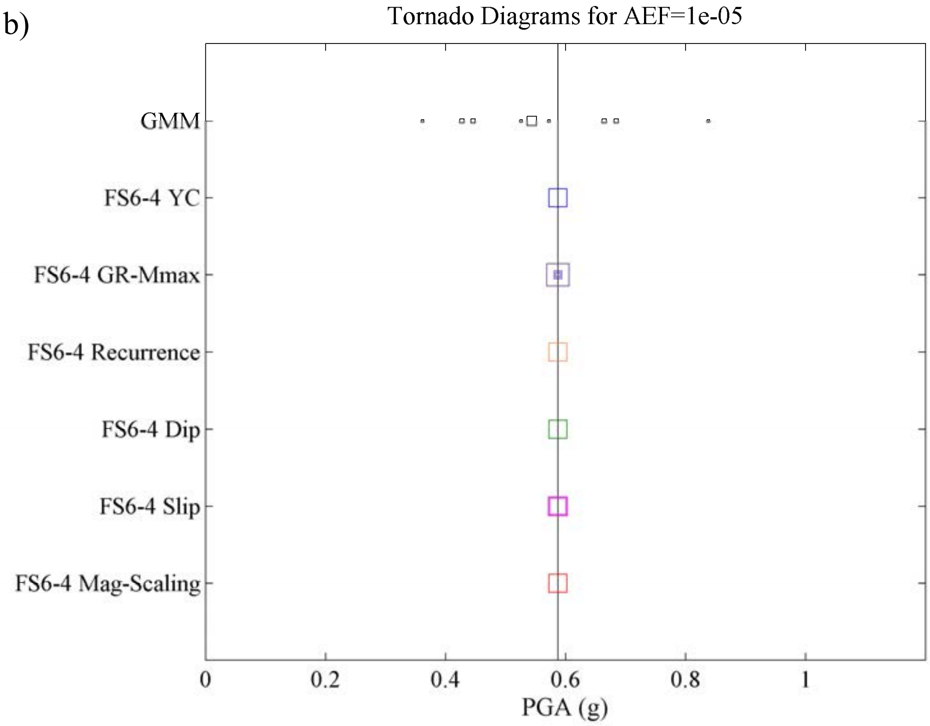
The investigated parameters in Tornado plots of FS6-4 are:

(1) GMM: The effect of the ground motion model, the median SA model and sigma model pairs defined in the GMC part of the study (9 branches).
(2) Dip: The effect of dip angle for the source designated in the prefix.
(3) Slip: The effect of slip rate for the source designated in the prefix.
(4) Mag Recurrence: The effect of magnitude recurrence model; i.e. the effect of GR truncated exponential and the characteristic recurrence model for the source designated in the prefix.
(5) GR-M_{max}: The effect of M_{max} selected for the branches modelled with GR truncated exponential magnitude distribution for the source designated in the prefix.
(6) YC: The effect of area rupture scaling law for the branches modelled with the characteristic model [II–13] and the magnitude-rupture area model [II–22].

a)



b)



c)

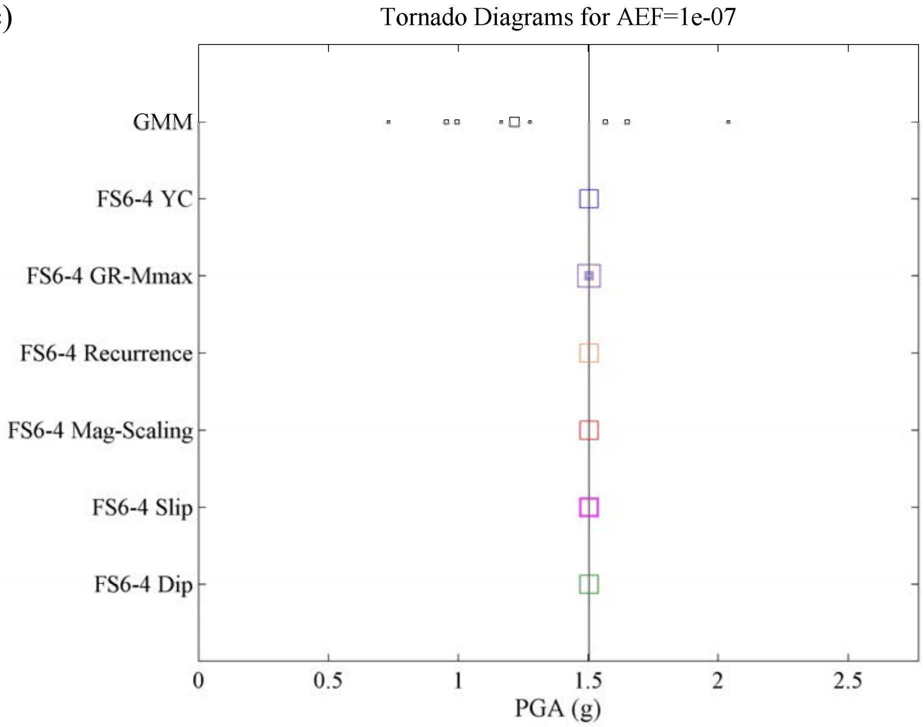
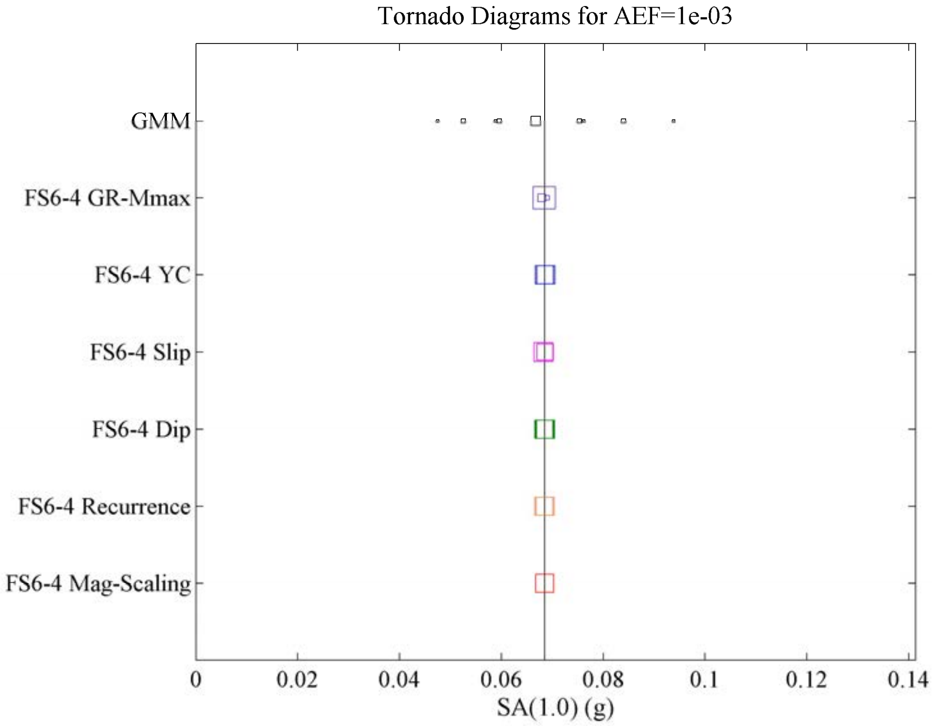
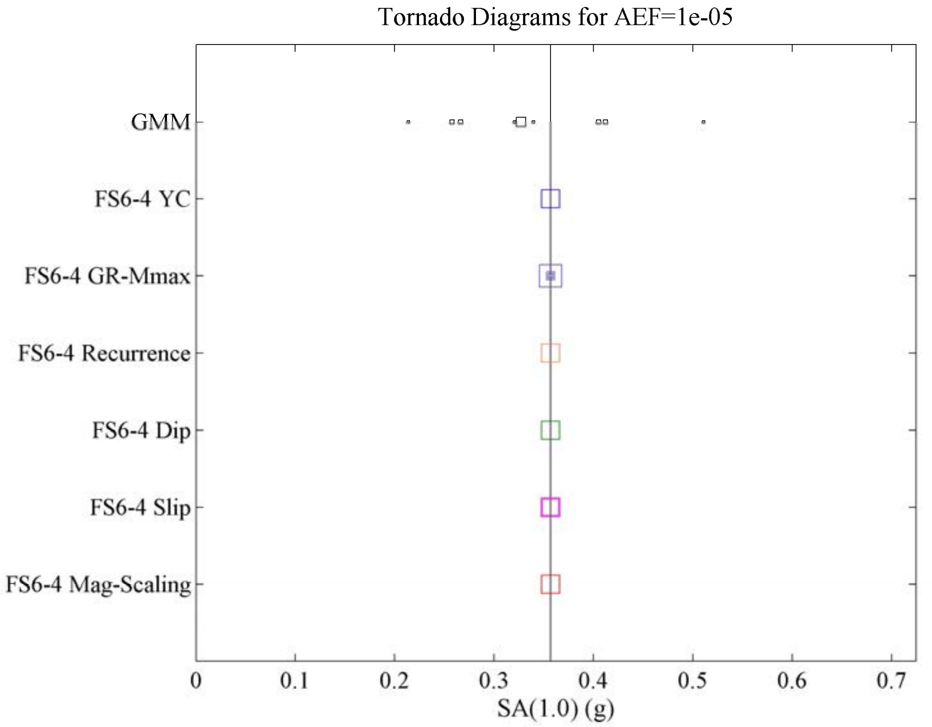


FIG. II–23. Tornado Plots of the fault source FS6-4 in the FSM1 model at AFEs equal to 10^{-3}, 10^{-5}, and 10^{-7} for PGA.

a)


Tornado Diagrams for AEF=1e-03
GMM
FS6-4 GR-Mmax
FS6-4 YC
FS6-4 Slip
FS6-4 Dip
FS6-4 Recurrence
FS6-4 Mag-Scaling
0
0.02
0.04
0.06
0.08
0.1
0.12
0.14
SA(1.0) (g)

b)


Tornado Diagrams for AEF=1e-05
GMM
FS6-4 YC
FS6-4 GR-Mmax
FS6-4 Recurrence
FS6-4 Dip
FS6-4 Slip
FS6-4 Mag-Scaling
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
SA(1.0) (g)

c)

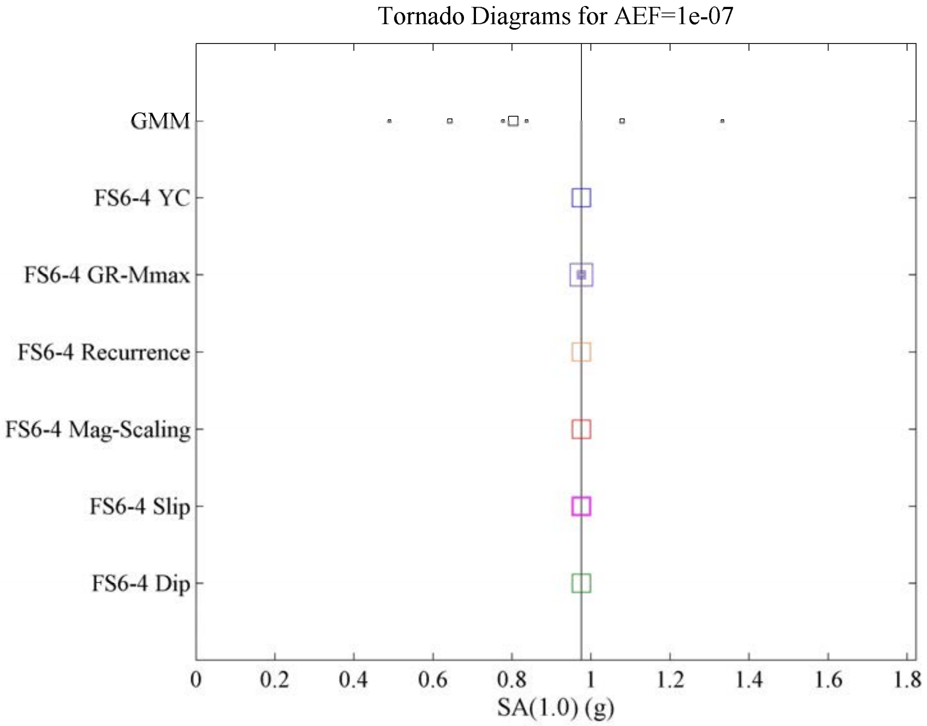


FIG. II–24. Tornado Plots of the fault source FS6-4 in the FSM1 model at AFEs equal to 10^{-3}, 10^{-5}, and 10^{-7} for T=1.0.

The following points summarize the observations from the comparative hazard results in this section:

(1) If alternative areal source models exhibit similar source geometry and activity rates, the uncertainty in their M_{max} would prevail. If the M_{max} values of such areal sources are very close to each other, the seismic source characterization team and its lead may consider disregarding the multiple alternatives and use the most critical one as the representative areal source model in seismic source characterization.

(2) The segmentation of fault sources is important for fault models. However, consideration of segmented/unsegmented alternatives of a distant fault from the SMR site is not important as the contribution of such alternatives are not significant to the PSHA results.

(3) The attempt of reflecting the CBR in fault source models would yield insignificant effect on the total hazard. Therefore, alternative branches of such fault sources can be disregarded in the PSHA model for simplification.

(4) The ground motion model and its epistemic uncertainty would dominate the total mean hazard.

II-8.4.3. Comparisons of the graded approach probabilistic seismic hazard analysis results with Full-Scale probabilistic seismic hazard analysis modelling

The PSHA model used in the application of a graded approach is based on a circular area of 50 km radius. The seismic source characterization and GMC models account for the epistemic uncertainties of each model parameter enclosed within this region although the separate observations on fault and areal source modelling advocate the possibility of further simplification in these parameters. (See discussions in previous two subsections). The full

PSHA model of the same example is much more comprehensive and encloses the areal and fault sources within the 300 km radius circular area.

The full seismic source characterization model is composed of fault and area source models that encompass the 300 km circular area. The alternative areal and fault models account for the epistemic uncertainties in several seismic domains that show different tectonic features within 300 km circular area. Therefore, the numbers of alternative areal and fault source models are greater than that are considered in the graded approach. As the area of interest in the full PSHA model is larger (300 km radius vs. 50 km radius), the number of seismic sources in each alternative fault and areal source model is significantly larger than that of the graded approach. Needless to say, the fault segments are longer having more complicated geometrical properties in the full seismic source characterization model. The consideration of multiple earthquake catalogues to account for the epistemic uncertainty in recurrence model parameters, consideration of different modes of hypocentral depth distribution, consideration of different modes of M_{max} distribution as well as the uncertainties in styles-of-faulting, fault slip rates result in a 10 times larger number of logic tree branches with respect to the one in the graded approach to the seismic source characterization model.

The GMC modelling of full scale PSHA compiles a project-specific ground motion database to be used while ranking the compiled GMMs through several data driven statistical testing methods (e.g. likelihood or log-likelihood testing methods). The ranking results are used to decide on the final set of GMMs that are used in the GMC logic tree framework. The full GMC modelling assigns specific weights to each one of the selected GMMs in the GMC logic tree. This process is cumbersome and lengthier than the scaled backbone approach described for the graded approach because the weighting strategies change for different magnitude-distance intervals in the GMC logic tree for a full coverage of CBR in spectral amplitudes. In essence, although the populated logic tree branch number is comparable between the full GMC model and the one proposed in the graded approach, the complexity in the implementation of full GMC modelling to PSHA slows down the computations significantly.

The above summary advocates significant differences between the two models. The full PSHA model is tailored at the level of a typical SSHAC Level 3 project. The proposed procedure is less comprehensive although the full CBR is intended to cover for both seismic source characterization and GMC model parameters within the 50 km radius encompassing the SMR site.

Figures II–25 and II–26 compare the 16th percentile, 50th (median) percentile, 84th percentile, and mean PGA and SA at T=0.2 s hazard curves computed from the graded approach and full PSHA modelling. These two spectral ordinates can represent very short and short period spectral ordinate ranges used in designing the critical system and structural components in SMRs. The limited comparisons indicate either very similar or conservative hazard curves derived from the application of a graded approach. One particular explanation is that the simplifications in the seismic source characterization modelling, as well as the backbone approach implemented in GMC, lead to more conservative results compared to the full PSHA approach. The differences in the GMC modelling of graded and non-graded approaches could be the major sources of observed differences. In particular, the different modes of ground motion modelling for median and standard deviations affect the CBR coverage by these two approaches, which is also highlighted in the Tornado diagrams.

In this example, conservatism is assured only for the seismic source characterization modelling. This is necessary because the simplification is made on the seismic source characterization model only. By reducing the 300 kms to 50 kms, seismic sources are reduced and therefore, it is necessary to ensure that: (i) the sources in the reduced region still account for a very large percentage of the contribution to hazard for all structural frequencies and annual frequencies of exceedance (in this case more than 90%), (ii) all the approximations made in both the areal and fault sources within the reduced region are more than sufficient to recover the deficit of the few percentage points of the hazard. These approximations are conservative and mainly involve combining of sources using enveloping parameters.

Regarding the GMC modelling, there were two options. The first was to use exactly what was used for the full PSHA. However, this process is cumbersome because the weighting strategies change for different magnitude-distance intervals in the GMC logic tree for a full coverage of CBR in spectral amplitudes. Therefore, in order to take advantage of the reduced region, within which both distance and magnitudes are much less variable resulting in fewer magnitude-distance intervals, it was decided to also simplify the GMC model. This simplification may impact the hazard results differently, i.e. hazard results may also decrease. However, this is due to the general decrease of variability in the ground motion parameters in the reduced region and it is fully justified.

The purpose of the graded approach is to use a methodology that is commensurate with the radiological risks posed by the nuclear installation. The first step is to demonstrate that the SMR design that is under study can be considered using a graded approach and that it is appropriate to use a simplified approach for the PSHA. The simplification chosen for this study is mainly related to time and human resources to be devoted to the PSHA. To this end, the simplifications made in the seismic source characterization and GMC models are described throughout the text.

The time and human resources for the full (non-graded) approach is estimated on the basis of recently performed PSHAs for nuclear power plants. In several PSHA studies, the SSHAC approach has been implemented using a Level 3 or approaching this level. Therefore, the comparison of the simplified approach will be compared to a SSHAC Level 3 effort in terms of time and human resources.

The overall project duration of the graded approach is expected to be no more than 1–1.5 years which is significantly shorter than that of a regular SSHAC Level 3 PSHA project. Recent examples show that the latter PSHA study (SSHAC Level 3) generally lasts for about 3.5–4 years. In a regular SSHAC Level 3 project, the key members of seismic source characterization and GMC are composed of five to six members whereas the approach presented in this document requires half of this number. The PSHA modelling team is generally composed of two or three members in SSHAC Level 3 projects due to the complexity of models but a qualified PSHA modeller is sufficient for the graded approach as both the seismic source characterization and GMC models as well as the logic tree structure is relatively simple. The technical experts assisting the seismic source characterization and GMC teams at various modelling steps (referred to as proponent experts or technical facilitators in SSHAC terminology) are fewer (two or three persons in total) when compared to a regular SSHAC Level 3 project. The estimate of 1–1.5 years is conservative and a result of a limited simplification. Applying other simplifications may well limit this effort to below one year.

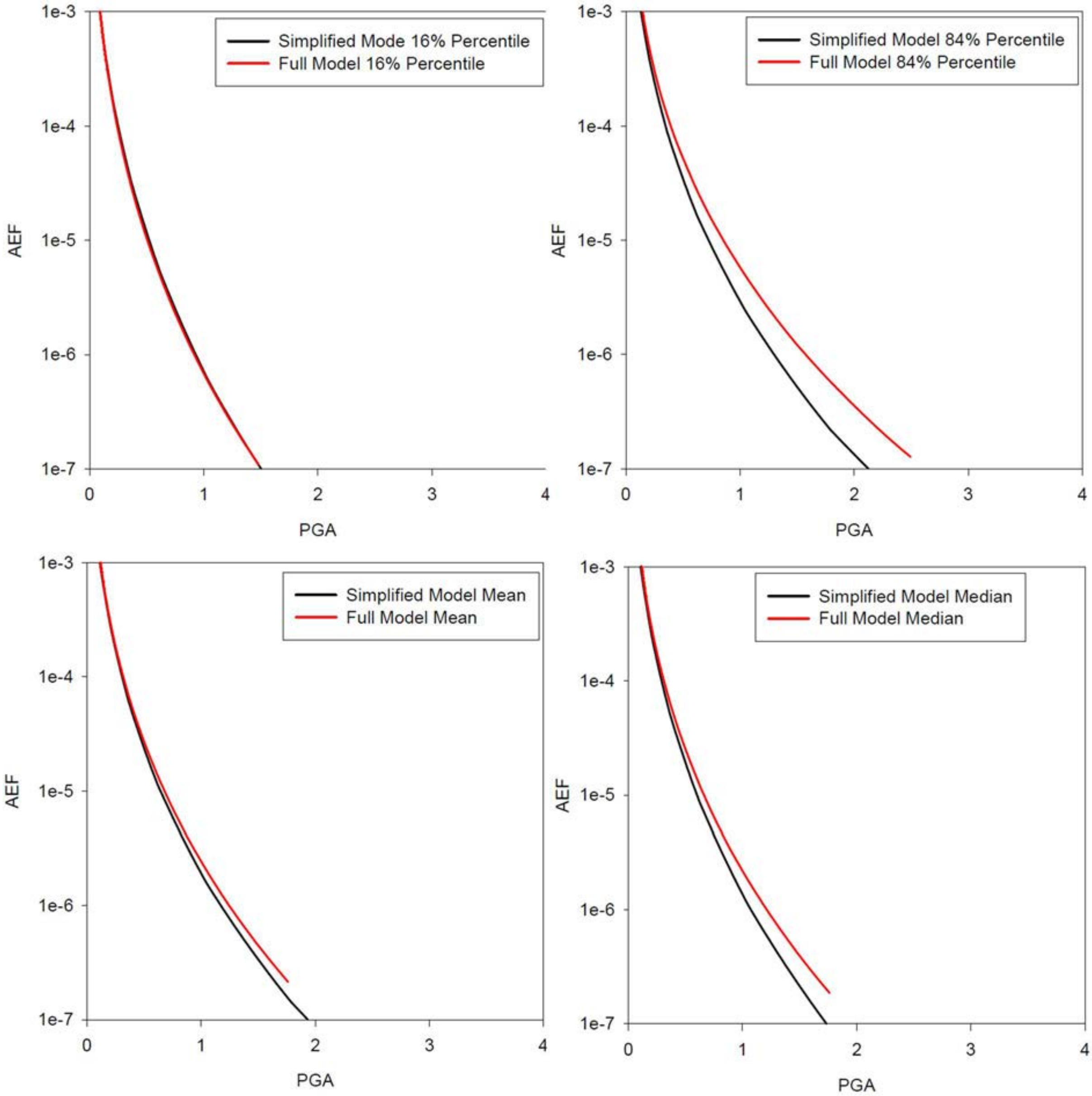


FIG. II–25. Comparisons of full PSHA model and the graded approach for PGA hazard for 16th percentile (upper left), 84th percentile (upper right), mean (lower left), and 50th percentile (median; lower right).

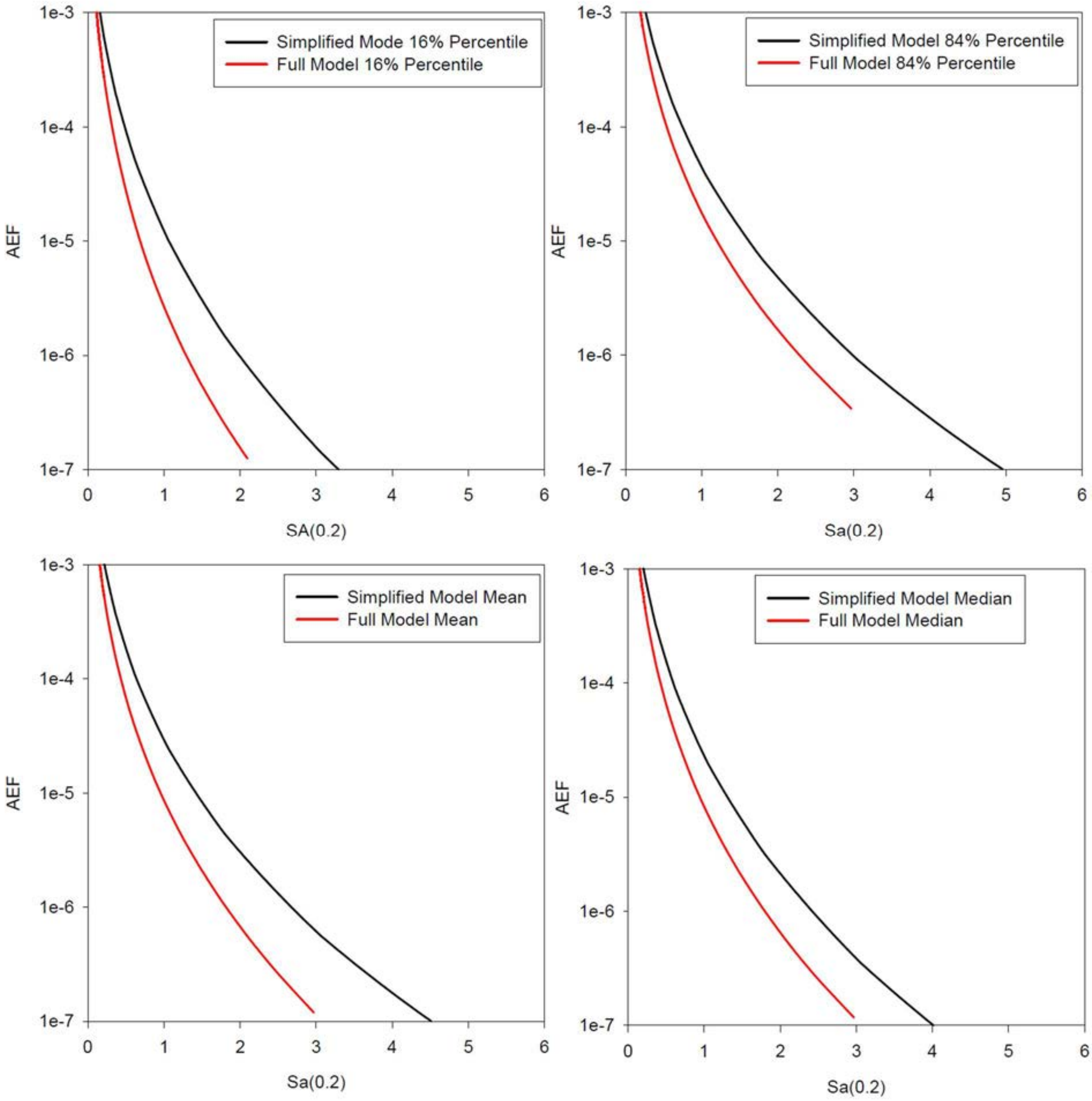


FIG. II–26. Comparisons of SSHAC Level 3 PSHA model and the graded approach for SA at T = 0.2 s hazard for 16th percentile (upper left), 84th percentile (upper right), mean (lower left), and 50th percentile (median; lower right).

II–9. CONCLUSIONS

A simplified graded approach methodology is proposed for the site specific PSHA of SMR sites. The approach focuses on the CBR of seismic source and ground motion characterization for the seismic sources that are located within about 50 km of the SMR site. The seismic source characterization lead and the team will determine the actual cut-off distance from deaggregation analysis of preliminary PSHA. The ground motion modelling considers the scaled backbone approach to address the CBR in median spectral amplitudes. The scaled backbone model tends to yield conservative median spectral amplitudes as the scaling functional form disregards the correlation between the predictor parameters (such as M and R).

A methodology similar to the modelling of median SA is used for developing the sigma model. The proposed approach yields a GMC logic tree framework that complies with the mutually exclusive and collectively exhaustive logic tree branches. Consequently, the weights in the logic tree framework are transformed into probabilities while computing different percentiles of vibratory ground motion hazard distribution. The GMC considers spectral periods up to T = 1.0 s (starting with the PGA) as longer spectral period demand is not needed for design and/or probabilistic safety analysis of SMRs unless there are additional technical concerns.

The seismic source characterization modelling establishes a comprehensive seismic source characterization logic tree framework after considering all possible modelling uncertainties in areal and fault sources within the radial area (designated as R = 50 km in this study). The comparisons of PSHA results as well as the Tornado diagrams suggest that further simplifications are possible in seismic source characterization modelling without producing doubts about the reliability of the PSHA results. As such, alternative areal source models having similar activity rates and geometries can be transformed into a single model unless their M_{max} values differ significantly. Such a simplification has insignificant effects for AFEs down to 10^{-7}. The epistemic uncertainty in fault source modelling is only important if the fault sources are close to the SMR site. Consideration of epistemic uncertainty of remotely located fault sources in the seismic source characterization logic tree only complicates and lengthens the PSHA calculations. The ground motion model and its associated uncertainty have a major impact on the hazard at the SMR site.

Limited comparisons are done between the hazard results from the application of a graded approach and the full PSHA model that is carried out by following the SSHAC Level 3 standards [II–5, II–6]. The comparisons are done for different percentiles of PGA and SA at T = 0.2 s (5 Hz spectral ordinate) hazard curves. The hazard curves computed from the graded approach yield either similar or larger spectral ordinates with respect to those of the full PSHA. The conservatism of a graded approach while modelling the uncertainties in PSHA can be an explanation to this observation. However, by carrying out more enhanced comparative PSHA studies associated with a more careful control over the graded and full PSHA modelling would lead to more comprehensive and better constrained explanations on the observed differences.

II–10. POTENTIAL IMPROVEMENTS TO THE SIMPLIFIED APPROACH

PSHA is a complex process which is also evolving rapidly. The present example used only some aspects of PSHA as a target for simplification. While the simplification that was chosen is very significant in terms of saving resources, there may be other simplifications that could be analysed in the future. Furthermore, in the beginning some assumptions were made in order to provide boundaries to the PSHA calculations. In future studies, it may be possible to relax some of these constraints in order to make the approach applicable to a wider range of sites.

Below are a few subjects that may be explored for potential improvements to the simplified approach in the future:

- SSG-9 (Rev. 1) [II–2] recommends the compilation of a project-specific earthquake catalogue which includes paleoseismolocial and archaeo-seismological data. It may also be necessary to conduct earthquake specific historical research (for historical events with large uncertainties). In general, these data are used for estimating M_{max} values. If M_{max} is calculated conservatively using seismic source characterization analogues then maybe lengthy research into paleo, archaeo and event-specific historical seismological work may be avoided.

- If there is literature that supports the existence of a major fault within the 50-60 km region of the site with no reliable seismicity associated with it, then a similar approach to the above may be adopted instead of lengthy geophysical investigations, provided that the fault is not too near the site having fault capability implications.
- What has been analysed in the present example represents a common seismotectonic situation for an NPP site. There will always be a small to moderate seismic source within, for example, the 50 km radius of the site whose contribution to seismic hazard would be significant and which needs to be quantified. However, there will be cases where additional and more defined sources (such as a subduction zone) will also have a significant contribution. Future work can be directed to the simplification of more complex seismic source configurations.
- It has been assumed that the site response can be handled in the PSHA using the regression parameters of the GMPEs. Simplified methods may be investigated which may allow similar approaches to be used for sites with more pronounced site effects (however, still avoiding sites which are susceptible to soil failures).
- The uncertainty in reference site conditions may be considered in the GMC logic tree.
- Effect of considering partially non-ergodic sigma in ground-motion modelling on reducing the ground-motion amplitudes for AFEs down to 10^{-6} may be studied.
- More example PSHAs with other source modelling features (e.g. smoothed seismicity approaches) may be performed.

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ANNEX III.
EXAMPLE OF NUCLEAR INSTALLATION AND SITE PARAMETERS NEEDED FOR SELECTION OF A SUITABLE SITE

Table III–1 presents typically used plant parameters and Table III–2 presents an example of parameters related with site characteristics for selection of a suitable site. Table III–3 provides typical exclusion criteria to be used for the site selection process.

TABLE III–1. TYPICAL NUCLEAR INSTALLATION CHARACTERISTICS FOR SITE SELECTION

No.	nuclear installation Parameter	Description
1	Installation type	Brief Description of Installation
2	Power (thermal and Electrical)	Thermal and Electrical power in Mw/h
3	Fuel type	Short Description
4	Onsite/Off Site Electrical Power	Short Description
5	Active safety systems	Description, demand of support systems, redundancy and diversity
6	Passive Safety Systems	Description, time of survival without support systems and operator actions
7	Ventilation systems	Description, temperature limits
8	Demand of cooling water (Safety related and process related)	for safety, for condenser cooling if applicable, temperature limits
9	Cooling Towers	characteristics
10	Ponds	Volume
11	Ultimate Heat Sink	Description
12	Containment system,	Description including Design Extension Conditions (DEC) functions
13	Water discharge	Volume, and temperature
14	Radiological releases and Source Term	Including χ/Q for accident conditions (concentration the radionuclide for a specific location and release height over Release Rate)
15	Design Basis Earthquake	GMRS, PGA
16	Design Basis air temperature	Annual Mean Max and extreme values
17	Design Basis precipitation	Extreme values
18	Design Basis Wind	Extreme values
19	Design Basis explosion	e.g. nature and mass of explosive material, distance, type of accident
20	Design Basis aircraft crash	e.g. mass and speed at impact, aircraft dimensions, altitude at impact, amount of fuel
21	Margins against hazards	Relative to bounding parameters
22	Protection against aircraft crash	Description
23	Protection against fire and explosion	Description

TABLE III–2 SITE CHARACTERISTICS

No.	Site Parameter	Description
1	Site type	e.g. Green, Brown or Existing NPP site and availability of site related information
2	Land characteristics availability	Surface, shape
3	Existing facilities on the site	Description
4	Geology, Seismology and Geotechnics	Description
5	Hydrology	Description and relevant parameters such as site elevation above the water body
6	Meteorology	Air temperature, precipitation and wind characteristics
7	Human Induced Hazards	Description including near-site industrial facilities, transportation risk, etc.
8	Atmospheric Dispersion characteristics	Description
9	Site and region infrastructure	Description
10	Population distribution	Description
11	Feasibility of Emergency Planning	Description
12	Environmental Characteristics	Environmental constraints

TABLE III–3 SITE EXCLUSIONARY CRITERIA

No.	Exclusionary Criteria	Description
1	Feasibility of implementation of emergency plan	Not feasible since timely evacuation and protective measures are not possible
2	Unacceptable Environmental Impact	Unacceptable Impact to reservations, flora and fauna etc.
3	Capable faults and potential permanent ground deformation within less than 5Km distance (or within site boundary)	Engineering protective solutions are not practical and the risk is too high
4	Massive Land Slides	Engineering protective solutions are not practical
5	Massive Liquefaction	Engineering protective solutions are not practical
6	Massive Karst	Engineering protective solutions are not practical
7	Lava Flow	No engineering protective solutions are available
8	Pyroclastic flow	No engineering protective solutions are available
9	Ground Deformation	No engineering protective solutions are available
10	Lahars (massive)	No engineering protective solutions are available
11	Significant exceedance of the design basis by some site specific hazards parameters	Engineering protective solutions are not practical

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24-00417E

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